brackets or parentheses . . .'' to accommodate the editing style of the various States.

Comments have not been received from the remaining groups.

Status of Licensee Metrication Efforts

Reactors

Although there are no power reactor licensees operating in the metric system, some of the advanced reactors have vendor-generated licensing documents that use the metric system of measurement. For example, both of General Electric's applications for the ABWR and SBWR designs have their Standard Safety Analysis Reports (SSAR) in the SI system of measurement. However, both the Westinghouse AP600 and the ABB–CE System 80+ have their SSARs in the traditional inch-pound system. The NRC's completed Final Safety Evaluation Reports (FSER) for the System 80+ and the ABWR are in dual units as prescribed by the Commission's policy statement. When the FSERs for the AP600 and the SBWR are published, they also will be in dual units.

Selected Examples of Metric Usage

There are varying degrees of use of the metric system of measurement by the non-power reactor nuclear industries. Also, within a particular profession or industry, there are varying degrees of metric use. For example, in the field of radiation oncology, the centigray (an SI unit) has been the meter of therapy doses, while the millicurie and curie (traditional units) are used as the measure expressing quantity or dosages.

Health Physics

It is also the case that most of the operational health physics community still uses the traditional system of measurement because of the use of instrumentation that is calibrated or expressed in that system. Some newer instrumentation that offers dual-unit options will assist in metric conversion, as the new instruments are being integrated into existing stock.

Public Comment

The NRC staff, through this request, is inviting comment from interested individuals on the NRC's metrication efforts to learn if there is a need for the Commission to revise its metrication policy.

Electronic Access

Comments may be submitted electronically, in either ASCII text or Wordperfect format (version 5.1 or later), by calling the NRC Electronic Bulletin Board on FedWorld. The bulletin board may be accessed using a personal computer, a modem, and one of the commonly available communications software packages, or directly via Internet.

If using a personal computer and modem, the NRC subsystem on FedWorld can be accessed directly by dialing the toll free number: 1–800–303–9672. Communication software parameters should be set as follows: Parity to none, data bits to 8, and stop bits to 1 (N,8,1). Using ANSI or VT–100 terminal emulation, the NRC rulemaking subsystems can then be accessed by selecting the ''Rules Menu'' option from the ''NRC Main Menu.'' For further information about options available for NRC at FedWorld consult the ''Help/Information Center'' from the ''NRC Main Menu.'' Users will find the ''FedWorld Online User's Guides'' particularly helpful. Many NRC subsystems and databases also have a ''Help/Information Center'' option that is tailored to the particular subsystem.

The NRC subsystem on FedWorld can also be accessed by a direct dial phone number for the main FedWorld BBS: 703–321–8020; Telnet via Internet: fedworld.gov (192.239.93.3); File Transfer Protocol (FTP) via Internet: ftp.fedworld.gov (192.239.92.205); and World Wide Web using: http://www.fedworld.gov (this is the Uniform Resource Locator (URL)). If using a method other than the toll free number to contact FedWorld, then the NRC subsystem will be accessed from the main FedWorld menu by selecting the ''F—Regulatory, Government Administration and State Systems'', then selecting ''A—Regulatory Information Mall''. At that point, a menu will be displayed that has an option ''A—U.S. Nuclear Regulatory Commission'' that will take you to the NRC Online main menu. You can also go directly to the NRC Online area by typing ''/go nrc'' at a FedWorld command line. If you access NRC from FedWorld's main menu, then you may return to FedWorld by selecting the ''Return to FedWorld'' option from the NRC Online Main Menu. However, if you access NRC at FedWorld by using NRC's toll-free number, you will have full access to all NRC systems but you will not have access to the main FedWorld system. For more information on NRC bulletin boards call Mr. Arthur Davis, Systems Integration and Development Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415–5780; e-mail AXD3@nrc.gov.

Lastly, the Act has a reporting requirement for Federal agencies to include an annual metric report as part of their annual budget submission to the Congress. The reporting requirement expires in the fiscal year after an agency has fully implemented metric usage. Unless the Commission receives comment which would require it to revise its policy, it will consider its policy final and its conversion to the metric system complete.

Dated at Rockville, Maryland this 14th day of September 1995.

For the Nuclear Regulatory Commission.

James M. Taylor,

Executive Director for Operations.

[FR Doc. 95–23932 Filed 9–26–95; 8:45 am]

BILLING CODE 7590–01–P

Biweekly Notice

Applications and Amendments to Facility Operating LicensesInvolving No Significant Hazards Considerations

I. Background

Pursuant to Public Law 97-415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97-415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from August 30, 1995, through September 15, 1995. The last biweekly notice was published on Wednesday, September 13, 1995 (60 FR 47613).

Notice Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the Federal Register a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this Federal Register notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By October 27, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

Carolina Power & Light Company, et al., Docket No. 50-324, Brunswick Steam Electric Plant, Unit 2, Brunswick County, North Carolina

Date of amendment request: August 4, 1995

Description of amendment request: The proposed amendment will allow the loading and use of GE13 fuel assemblies in the Brunswick Steam Electric Plant (BSEP), Unit 2, during Cycle 12 operation. The use of GE13 fuel assemblies requires that the safety limit value for minimum critical power ratio be revised. This safety limit is established to maintain fuel cladding integrity. Use of GE13 fuel also requires an increase in the concentration of sodium pentaborate solution required by the Technical Specifications (TS) for the standby liquid control system. This change provides the additional shutdown reactivity necessary to permit use of this fuel.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Proposed Change 1:

The proposed amendment will allow the loading and use of GE13 fuel assemblies in the Brunswick Unit 2 reactor core. The use of GE13 fuel assemblies requires that the safety limit minimum critical power ratio value also be revised. The safety limit minimum critical power ratio is established to maintain fuel cladding integrity. The GE13 fuel assembly design has been analyzed using methods that have been previously approved by the Nuclear Regulatory Commission and documented in General Electric Nuclear Energy's reload licensing methodology Topical Report (NEDE-24011-P-A-10, "General Electric Standard Application for Reactor Fuel (GESTAR II)" dated February 1991).

The proposed revision of the safety limit minimum critical power ratio does not alter any plant safety-related equipment, safety function, or plant operations that could change the probability of an accident. The change does not affect the design, materials, or construction standards applicable to the fuel bundles in a manner that could change the probability of an accident.

A methodology that has been previously reviewed and accepted by the Nuclear Regulatory Commission was used to derive the both existing and updated safety limit minimum critical power ratio value. The same methodology criteria have been applied to derive the existing safety limit minimum critical power ratio of 1.07 as that used to derive the updated safety limit minimum critical power ratio value of 1.09. The updated safety limit minimum critical power ratio assures that fuel cladding protection equivalent to that provided with the existing safety limit minimum critical power ratio value is maintained. This ensures that the consequences of previously evaluated accidents are not significantly increased.

Proposed Change 2:

The standby liquid control system provides a means of reactivity control that is independent of the normal reactivity control system. The standby liquid control system must be capable of assuring that the reactor core can be placed in a subcritical condition at any time during reactor core life. Technical Specification Figure 3.1.5-1 specifies the acceptable range of concentrations and volumes for sodium pentaborate solution used as a neutron absorber (i.e., for reactivity control). The portion of the sodium pentaborate concentration range shown in Technical Specification Figure 3.1.5-1 applicable to the lower range of tank volumes is being revised to increase the required concentration of sodium pentaborate solution. This change is needed to account for the additional shutdown reactivity needed based on the planned use of GE13 fuel assemblies as reload fuel for the Unit 2 reactor core. Since the standby liquid control system is independent from the normal means of controlling reactor core reactivity and not used to control core reactivity during normal plant operations, the proposed revision to the sodium pentaborate concentration curve for the standby liquid control system does not alter any plant safety-related equipment, safety function, or plant operations that could change the probability of an accident.

The current volume-concentration range of sodium pentaborate used in the standby liquid control system will achieve a sufficient concentration of boron in the reactor vessel to ensure reactor shutdown. Based on the increased reactivity of the new GE13 reload fuel assemblies, the required sodium pentaborate volume-concentration range is being revised to ensure sufficient neutron absorbing solution is available to achieve reactor shutdown; therefore, the consequences of an accident previously evaluated are not significantly increased.

2. The proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

Proposed Change 1:

The GE13 fuel assembly has been designed and complies with the acceptance criteria contained in General Electric Nuclear Energy's standard application for reactor fuel (GESTAR-II), which provides the latest acceptance criteria for new General Electric fuel designs. The GE13 fuel assembly complies with GESTAR-II acceptance criteria that have been previously reviewed and accepted by the Nuclear Regulatory Commission. The similarity of the GE13 fuel design to the previously accepted GE11 fuel design, in conjunction with the increased critical power capability of the GE13 fuel design, ensure that no new mode or condition of plant operation is being authorized by the loading and use of the GE13 fuel type. The proposed revision of the safety limit minimum critical power ratio from 1.07 to 1.09 does not modify any plant controls or equipment that will change the plant's responses to any accident or transient as given in any current analysis. Therefore, the proposed change to allow the loading and use of the GE13 fuel type and the revision of the safety limit minimum critical power ratio value from 1.07 to 1.09 will not create the possibility for a new or different kind of accident from any accident previously evaluated.

Proposed Change 2:

As discussed above, the standby liquid control system provides a means of reactivity control that is independent of the normal reactivity control system and is capable of assuring that the reactor core can be placed in a subcritical condition at any time during reactor core life. The proposed revision to the sodium pentaborate concentration range does not modify the standby liquid control system or its controls, does not modify other plant systems and equipment, and does not permit a new or different mode of plant operation. As such, the proposed revision to the minimum pentaborate concentration value does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license amendment does not involve a significant reduction in a margin of safety.

Proposed Change 1:

As previously discussed, the GE13 fuel assembly design has been analyzed using methods that have been previously approved by the Nuclear Regulatory Commission and documented in General Electric Nuclear Energy's reload licensing methodology Topical Report (NEDE-24011-P-A-10, "General Electric Standard Application for Reactor Fuel (GESTAR II)" dated February 1991). The safety limit minimum critical power ratio value is selected to maintain the fuel cladding integrity safety limit (i.e., that 99.9 percent of all fuel rods in the core be expected to avoid boiling transition).

Appropriate operating limit minimum critical power ratio values are established, based on the safety limit minimum critical power ratio value, to ensure that the fuel cladding fuel integrity safety limit is maintained. The operating limit minimum critical power ratio values are incorporated in the Core Operating limits Report as required by Technical Specification 6.9.3.1. The new GE13 safety limit minimum critical power ratio value of 1.09 is based on the same fuel cladding integrity safety limit criteria at that for the GE11 safety limit minimum critical power ratio value of 1.07 (i.e., that 99.9 percent of all fuel rods in the core be expected to avoid boiling transition); therefore, the proposed change does not result in a significant reduction in the margin of safety.

Proposed Change 2:

As previously stated, the purpose of the standby liquid control is to inject a neutron absorbing solution into the reactor in the event that a sufficient number of control rods cannot be manually inserted to maintain subcriticality. Sufficient solution is to be injected such that the reactor will be brought from maximum rated power conditions to subcritical over the entire reactor temperature range from maximum operating to cold shutdown conditions. General Electric reactor fuel methodology establishes a fuel type dependent standby liquid control system shutdown margin to account for calculational uncertainties. General Electric calculations show that an in-vessel concentration of 660 ppm will provide an estimated standby liquid control system minimum shutdown margin of 4.1% delta k. To achieve an in-vessel concentration of 660 ppm, the acceptable range of standby liquid control system tank concentrations is being revised for the lower range of tank volumes. Thus, proposed revision of the standby liquid control system sodium pentaborate volume-concentration range ensures that there will not be a significant reduction in the amount of available shutdown margin and, therefore, not a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297.

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: David B. Matthews

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: April 10, 1995

Description of amendment request: The requested amendment would modify Technical Specification 4.6.4.3 to allow a reduction in the number of hydrogen mitigation system igniters that must be maintained Operable. This would allow removal of the hydrogen igniters in the incore instrument tunnel.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1

The requested amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated. No impact upon accident probabilities will be created, since the EHM System is not an accident initiating system. In addition, it has been demonstrated that based on the results of computer analysis, and the review of results of an external study performed for a similar type containment, that hydrogen concentrations in the cavity during degraded core accidents will remain within acceptable limits. No impact on the plant response to any accident will be created (either design basis or beyond-design basis).

Criterion 2

The requested amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. As stated previously, the EHM System is not an accident initiating system. No new accident causal mechanisms will be created as a result of deleting the affected igniters. Plant operation will not be affected by the proposed amendments and no new failure modes will be created.

Criterion 3

The requested amendments will not involve a significant reduction in a margin of safety. No adverse impact upon any plant safety margins will be created. As shown previously, applicable computer analysis has successfully demonstrated that the affected igniters could be removed with no adverse consequences. No fission product barriers are being degraded. No change to the manner in which the units are operated is being made.

Based upon the preceding analyses, Duke Power Company concludes that the requested amendments do not involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242

NRC Project Director: Herbert N. Berkow

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: September 1, 1995

Description of amendment request: Generic Letter 88-16 provided guidance on removing cycle-specific parameters which are calculated using NRC approved methodologies from Technical Specifications (TS). The parameters are replaced in TS with a reference to a named report which contains the parameters, and a requirement that the parameters remain within the limits specified in the report. The proposed changes incorporate NRC approved methodologies, approved revisions to previously approved methodologies, or republished versions of previously approved methodologies into Section 6.9 of the Catawba TS. For Catawba, the limits to which these methodologies are applied are explicitly listed in the TS. Since the proposed changes only incorporate NRC approved methodologies into the TS the licensee proposed that the changes are administrative in nature and can be assumed to have no impact, or potential impact, on the health and safety of the public.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed changes will not create a significant hazards consideration, as defined by 10 CRF 50.92, because:

1) The proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes are administrative in nature, and do not affect any system, procedure, or manipulation of any equipment which could affect the probability or consequences of any accident.

2) The proposed changes will not create the possibility of any new or different kind of accident from any accident previously evaluated.

The proposed changes are administrative in nature, and cannot introduce any new failure mode or transient which could create any accident.

3) The proposed changes will not involve a significant reduction in a margin of safety.

The proposed changes are administrative in nature, and will not affect any operating parameters or limits which could result in a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242

NRC Project Director: Herbert N. Berkow

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: September 13, 1995

Description of amendment request: The proposed amendments modify the notation for the overpower delta-temperature (OPDT) reactor trip heatup setpoint penalty coefficient to be consistent with NUREG-0452, Revision 4, ''Standard Technical Specifications for Westinghouse Pressurized Water Reactors'' (STS). This change is necessary in order to allow implementation of the modification to reduce the reactor coolant system hot leg temperature as planned during the Unit 2 end-of-cycle 7 refueling outage.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

As required by 10CFR50.91, this analysis is provided concerning whether the requested amendments involve significant hazards considerations, as defined by 10CFR50.92. Standards for determination that an amendment request involves no significant hazards considerations are if operation of the facility in accordance with the requested amendment would not: 1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or 2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) Involve a significant reduction in a margin of safety.

Criterion 1

The proposed amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated. The amendments will have no impact whatsoever upon the probability of any accident being initiated, since the reactor trip system is an accident mitigating system. The amendments will have no adverse impact upon any accident consequences or upon the function of the OPDT setpoint. The reactor trip heatup setpoint penalty will continue to be applied anytime T-avg is greater than T [double prime] and will not be applied when T-avg is less than or equal to T [double prime]. This is consistent with the intent of this function.

Criterion 2

The proposed amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. The function of the OPDT setpoint will not be altered by the proposed changes. As stated previously, the reactor trip system is an accident mitigating system, so no new failure modes can be created. No change to any aspect of plant operation will result from NRC approval of the proposed amendments.

Criterion 3

The proposed amendments will not involve a significant reduction in a margin of safety. The changes are necessary to allow full implementation of the T-hot reduction modification on Catawba Unit 2. The proposed changes are consistent with the terminology of both NUREG-0452, Revision 4 and NUREG-1431, Revision 1. OPDT setpoint behavior will not be adversely impacted by the proposed changes; therefore, no impact upon any plant safety margins will result.

Based upon the preceding analyses, Duke Power Company concludes that the requested amendments do not involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242

NRC Project Director: Herbert N. Berkow

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: March 29, 1995

Description of amendment request: The amendments would revise the Technical Specification 3.4.9.3 requirements for the Low Temperature Overpressure Protection (LTOP) system and update the heatup and cooldown curves. The intent of the proposed amendments is to enhance overpressure protection during low temperature operations. These enhancements can be fully implemented, improving startup and shutdown operation of McGuire Units 1 and 2.

Specifically, these changes are categorized into five groups identified as follows:

1) Revisions to the LCO requirements, the Action Statements and the SR for the Reactor Coolant System Overpressure Protection System during low temperature conditions,

2) A reduction in the Reactor Coolant System (RCS) vent requirement from 4.5 square inches to 2.75 square inches,

3) The use of the Residual Heat Removal suction relief valve (1ND3 and 2ND3) for overpressure protection under restricted conditions. (RCS greater than 107°F and cooldown rate less than 20°F/hr; or RCS greater than 167°F),

4) Revisions of the Pressure/Temperature curves to 16 EFPY, including the incorporation of the latest radiation surveillance capsule results and removal of instrumentation margins from the Technical Specification figures, and

5) Changes to format and consistency.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration for each of the five groups listed above.

FIRST STANDARD

(Amendment would not) involve a significant increase in the probablility or consequences of an accident previously evaluated.

1) Revised LCO [limiting conditions for operation] and SR [surveillance requirements] for LTOP:

The reduced maximum setpoint will prevent the violation of the 10 CFR 50 Appendix G pressure/temperature curves (as modified by the provisions of ASME Code Case N-514) during overpressure transients at low temperatures. Since the maximum setpoint is reduced, the peak pressure for LTOP [low-temperature overpressure protection] events will be reduced as well. Accordingly, the consequences of an LTOP event would not change as result of the proposed changes.

The analysis performed to determine the setpoint is, in accordance with the methods used in previous evaluations, found acceptable by the NRC. The three possible transients evaluated are; 1) a mass input from an operable safety injection pump; 2) a mass input from an operable centrifugal charging pump; and 3) a heat input from a 50°F temperature difference between the steam generators and the NC system. The LTOP setpoint of the PORV [power-operated relief valve] proposed by this technical specification change is not considered to be an initiator of any of these three transients. As such, the probability of an accident previously evaluated would not be increased as a result of the proposed changes.

Two additional conditions for operability of the LTOP system are defined (accumulator isolation and only one NV or NI pump operable) and new surveillance requirements are specified as well. They provide additional limitations, requirements and restrictions that currently do not exist within the technical specifications for McGuire. The incorporation of these proposed changes are consistent with what is specified within NUREG-1341. Therefore, these changes do not increase the probability of consequences of an accident previously evaluated.

2) Reduction in NC vent opening:

The bases for the size of the vent to be established per the technical specifications is to ensure that the 10 CFR 50, Appendix G pressure/temperature limits are not exceeded during an LTOP event. The determination of the size of the opening continues to preserve the above design basis. The evaluation performed demonstrated that a 2.75 square inch opening would provide adequate overpressure protection for the combined capacity of a centrifugal charging pump and a safety injection pump.

The only time that the vent path is to be established is when the PORVs may not be available. Defining the size of the vent is not considered to be an initiator of any LTOP events that have been previously evaluated. As such, this change in the size of the vent opening does not increase the probability of an overpressure event during low temperature conditions. The analysis performed verifies that the size opening specified is sufficient to mitigate the consequences of an LTOP event. Accordingly, the change in the size of the opening for the vent will not impact the consequences of LTOP events.

3) Use of RHR [residual heat removal] suction relief valves:

By letter dated September 11, 1990, the NRC authorized the deletion of the RHR autoclosure interlock circuitry. A modification which removed the RHR system suction isolation valve autoclosure interlocks has been completed. As such, the RHR suction relief valve can be exposed to NC system pressure and would be available to mitigate LTOP events.

The proposed amendments specify the necessary requirements and controls to ensure proper ND system alignments and conditions will exist to protect the pressure/temperature limits. This added relieving capacity will enhance the current LTOP system at McGuire in mitigating overpressure events at low temperatures. As such, the mitigation of previously evaluated LTOP events would be improved by the proposed technical specification changes. Further, the proposed changes would not esult in the initiation of an LTOP event or cause an overpressure transient. Accordingly, the proposed amendment would not involve an increase in the consequences or the probability of an accident previously evaluated.

4) Revised pressure/temperature curves to 16 EFPY [effective full-power year]:

The proposed pressure/temperature curves, provided by this amendment request, satisfy all regulatory required material embrittlement considerations including: ASME Section XI Appendix G, 10 CFR 50 Appendix G, and Regulatory Guide 1.99, Revision 2. In addition, the margins for instrument error have been removed from the curves. Instrument error will be administratively handled by incorporating them into the LTOP system setpoint selection calculations and into appropriate controlling procedures for unit operations.

The proposed changes to the pressure/temperature curves are not considered to be an initiator of LTOP events. The changes to the curves proposed by this amendment request will not cause an LTOP event. The curves define the new limits that have been defined in accordance with regulatory requirements by which both units are to be operated within. Accordingly, the proposed amendment will not increase the probability or the consequences of previously evaluated accidents.

5) Format and consistency:

The changes associated within this group are considered to be administrative in nature. They do not affect station operability or require any modifications to the facility. Accordingly, the proposed amendment request does not increase the probability or consequences of any previously evaluated accident.

SECOND STANDARD

(Amendment would not) create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

1) Revised LCO and SR for LTOP:

The only potential impact to plant systems, structures and components, as a result of the proposed changes associated with this group, would be the setting of the PORV low pressure setpoint. No other changes to plant systems, structures or components would occur. The proposed amendments, also, would not impact the plant operation. Although the value for the PORV pressure setting specified within the technical specification would be reduced per the proposed amendment, the actual settings of the PORV are now currently set low enough to comply with the proposed lower setpoint value. As such, the proposed lower setpoint would not require any changes to the plant nor how the plant is operated.

The additional requirements for LTOP operability will not require any modifications to the plant nor how the plant is operated. Currently, when entering LTOP conditions, the accumulators are isolated and only one NV or NI pump is capable of injecting into the reactor vessel. these actions are currently controlled and are specified within the operating procedures for heatup and cooldown of the respective units. The proposed changes will now specify these current operating requirements within the technical specifications as well.

Accordingly, the proposed revisions will not create a new or different kind of accident than what has already been previously evaluated.

2) Reduction in NC vent opening:

The proposed changes to the technical specifications associated with this group involves the size of the vent opening. The proposed amendment reduces the size of the vent opening from 4.5 square inches to 2.75 square inches. The analysis that was performed has determined that the proposed size for the vent opening is adequate for overpressure events. Therefore, this proposed revision to the technical specifications will not result in a new or different kind of accident from any kind of accident previously evaluated.

3) Use of RHR suction relief valves;

The proposed amendment associated with this group will specify the necessary requirements and controls to ensure the appropriate use of the RHR suction relief valve for overpressure protection. This added relieving capacity will enhance the current LTOP system in mitigating overpressure events during low temperature conditions. The analysis that has been performed demonstrates the adequacy of the RHR suction relief valve, in conjunction with a PORV, in mitigating overpressure events at low temperatures, assuming a worst case single failure as well. As such, the use of the RHR suction relief valve in the manner prescribed by the proposed technical specification amendment will not create a new or different kind of accident from those accidents that have been previously evaluated.

4) Revised pressure/temperature curves to 16 EFPY:

The changes associated with this group, provide new heatup and cooldown curves for both Units 1 and 2, which will extend the service period from 10 EFPY to 16 EFPY and will remove the instrument error as well. The proposed [heatup] and cooldown curves were developed in accordance with all regulatory required material embrittlement criteria. Thus, operation of the units in accordance with the proposed new pressure/temperature curves will not create the possibility of a new or different kind of accident from those accident[s] that have been previously evaluated.

5) Format and consistency:

The changes associated within this group are considered to be administrative in nature. They do not affect station operability or require any modifications to the facility. Accordingly, the proposed amendment will create the possibility of a new or different kind of accident from that previously evaluated.

THIRD STANDARD

(Amendment would not) involve a significant reduction in a margin of safety.

1) Revised LCO and SR for LTOP:

This proposed change will reduce the maximum PORV setpoint such that, for LTOP events, the maximum pressure in the vessel would not exceed 110% of the pressure/temperature limits that have been established in accordance with ASME Appendix G. This is congruous with the provisions of ASME Code Case N-514. Currently, the maximum PORV setpoint for LTOP events ensure that the maximum pressure would not exceed 100% of the pressure/temperature curves. As such, the proposed change appears to involve a slight reduction in a margin of safety.

Although the proposed change may involve a slight reduction in a margin of safety, the proposed change will provide an equivalent margins of safety to the reactor vessel during LTOP transients and will satisfy the underlying purpose of 10 CFR 50.60 for fracture toughness requirements. By letter dated June 28, 1994, an exemption request and authorization to use ASME Code Case N-514 at McGuire was submitted to the NRC for review and approval. Approval for the use of the code case was granted on September 30, 1994. The proposed change to reduce the maximum PORV setpoint, coupled with the September 30, 1994 NRC approval for the use of Code Case N-514 satisfies current regulatory acceptance criteria. Therefore, the proposed change would not involve a significant reduction in a margin of safety.

This change group, also, defines two additional conditions for the operability of the LTOP system (accumulator isolation and only one NV or NI pump operable) and proposes new surveillance requirements and restrictions that currently do not exist within the technical specifications for McGuire. The incorporation of these proposed changes are consistent with what is specified within NUREG-1341. Therefore, these changes do not involve a significant reduction in a margin of safety.

2) Reduction in NC vent opening:

The proposed changes to the technical specifications associated with this group involves the size of the vent opening. The proposed amendment reduces the size of the vent opening from 4.5 square inches to 2.75 square inches. The basis for the size of the vent to be established per the technical specifications is to ensure that the 10 CFR 50, Appendix G pressure/temperature limits are not exceeded during an LTOP event. The determination of the size of the opening continues to preserve the above design basis. The evaluation performed demonstrated that a 2.75 square inch opening would provide adequate overpressure protection for the combined capacity of a centrifugal charging pump and a safety injection pump. Accordingly, the proposed changes would not involve a significant reduction in a margin of safety.

3) Use of RHR suction relief valves:

The proposed amendment associated with this group will specify the necessary requirements and controls to ensure the appropriate use of the RHR suction relief valves for overpressure protection. This added relieving capacity will enhance the current LTOP system in mitigating overpressure events during low temperature conditions. The analysis that has been performed demonstrates the adequacy of the RHR suction relief valve, in conjunction with a PORV, in mitigating overpressure events at low temperatures.

Further, by letter dated September 11, 1990, the NRC approved amendments to delete a portion of the surveillance requirements regarding periodic verification that the RHR suction isolation valves automatically close on a RCS [reactor coolant system] signal less than or equal to 560 psig. This action, in effect, authorizes the removal of the RHR autoclosure interlock circuitry. As discussed within the NRC Safety evaluation for the amendment, the Commission and industry have recognized the safety benefits of removing the ACI [automatic closure and interlock] circuitry from the RHR system to minimize, and thus reduce the risk associated with loss of decay heat removal events.

Therefore, the proposed amendments associated with this change group will not involve a significant reduction in a margin of safety.

4) Revised pressure/temperature curves to 16 EFPY:

The changes associated with this group provide new heatup and cooldown curves for both Units 1 and 2, which will extend the service period from 10 EFPY to 16 EFPY and will relocate the instrument error as well. The proposed pressure/temperature curves provided by this amendment request satisfy all regulatory required material embrittlement considerations including; ASME Section XI Appendix G, 10 CFR 50 Appendix G, and Regulatory Guide 1.99, Revision 2. The instrument error will be administratively handled by incorporating them into the LTOP system setpoint selection calculations and into the controlling procedures for unit operations.

The relocation of the instrument error to licensee controlled documents is consistent with the NRC actions proposed within NUREG-1431, new standard technical specifications for Westinghouse plants. As prescribed within NUREG-1431, the pressure/temperature limit curves are to be relocated to a licensee controlled document entitled ''Pressure Temperature Limit Report (PTLR)''. Changes to the heatup and cooldown curves would then be performed in accordance with 10 CFR 50.59 criteria. For the situation proposed by this amendment, updates and revisions of the instrument error associated with the pressure/temperature limit curves will be processed in a similar fashion. Thus, the proposed change to relocate the instrument error to licensee controlled documents is analogous with NRC acceptable practices.

Accordingly, the proposed changes will not reduce a margin of safety.

5) Format and consistency:

The changes associated within this group are considered to be administrative in nature. They do not affect station operability or require any modifications to the facility. Accordingly, there is no reduction in the margin of safety of the LTOP system due to the incorporation of these editorial/administrative changes.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242

NRC Project Director: Herbert N. Berkow

Florida Power and Light Company, et al., Docket Nos. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: June 21, 1995

Description of amendment request: The proposed amendments will revise the action statements for a single inoperable Emergency Diesel Generator (EDG), TS 3.8.1.1.b, to extend the allowed outage time (AOT) from 72 hours to 7 days, and permit a 10 day AOT to be used once per refueling cycle. This proposal is a result of a cooperative study by participating Combustion Engineering Owners Group members which concluded that the proposed AOT extension improves plant operational flexibility while adequately controlling overall plant risk.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments for St. Lucie Unit 1 and Unit 2 will extend the action completion/allowed outage time (AOT) for a single inoperable Emergency Diesel Generator (EDG) from 72 hours to 7 days, with provisions for a 10 day AOT once per refueling cycle. The EDGs are designed as backup AC power sources for essential safety systems in the event of a loss of offsite power. As such, the EDGs are not accident initiators, and an extended AOT to restore operability of an inoperable diesel generator would not increase the probability of occurrence of accidents previously analyzed.

The proposed technical specification revisions involve the AOT for a single inoperable EDG, and do not change the conditions, operating configuration, or minimum amount of operating equipment assumed in the plant safety analyses for accident mitigation. In addition, a Probability Safety Assessment (PSA) was performed to quantitatively assess the risk impact of the proposed amendment. The impact on the early radiological release probability for design basis events was also evaluated. It was concluded that the risk contribution from this proposed AOT is very small, and that the impact will be negligible.

Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not

create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendments will not change the physical plant or the modes of plant operation defined in either Facility License. The changes do not involve the addition or modification of equipment, nor do they alter the design of plant systems. Therefore, operation of either facility in accordance with its proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed amendments are designed to improve EDG reliability by providing flexibility in the scheduling and performance of preventive and corrective maintenance activities. The surveillance intervals or the operability requirements are not changed by the proposal; only the AOT for a single inoperable EDG will be extended. The proposed changes do not alter the basis for any technical specification that is related to the establishment of, or the maintenance of, a nuclear safety margin. Moreover, an integrated assessment of the risk impact of extending the AOT for a single inoperable EDG has determined that the risk contribution is very small and can be offset by improvements in EDG reliability. Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

Attorney for licensee: J. R. Newman, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

Florida Power and Light Company, et al., Docket Nos. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: June 21, 1995

Description of amendment request: The proposed amendments will revise TS 3.5.2 to allow up to 7 days to restore an inoperable Low Pressure Safety Injection train to operable status. This proposal is a result of a cooperative study by participating Combustion Engineering Owners Group members which concluded that an extension of the allowed outage time (AOT) from 72 hours to 7 days can improve plant operational flexibility and is risk beneficial.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments for St. Lucie Unit 1 and Unit 2 will extend the action completion/allowed outage time (AOT) for a single inoperable Low Pressure Safety Injection (LPSI) train from 72 hours to 7 days. A LPSI train is designed as a part of each Emergency Core Cooling System (ECCS) subsystem to supplement Safety Injection Tank (SIT) inventory during the early stages of mitigating a Design Basis Accident. As such, components of the LPSI system are not accident initiators, and an extended AOT to restore operability of an inoperable LPSI train would not increase the probability of occurrence of accidents previously analyzed.

The safety analyses for both St. Lucie Units demonstrate that ECCS performance acceptance criteria are satisfied with only one of the two redundant ECCS subsystems operating during the postulated Design Basis Accident. The proposed technical specification revisions involve the AOT for a single inoperable LPSI train, and do not change the conditions assumed for the minimum amount of operating equipment needed for accident mitigation. Therefore, the consequences of an accident previously evaluated will not be significantly increased.

In addition to the preceding evaluation, a Probabilistic Safety Analysis (PSA) was performed to quantitatively assess the risk impact of the proposed amendments. It was concluded from the results of that assessment that the risk contribution of the AOT extension is very small, and that the net impact of the proposed amendment can be risk beneficial.

Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendments will not change the physical plant or the modes of plant operation defined in either Facility License. The changes do not involve the addition or modification of equipment nor do they alter the design of plant systems. Therefore, operation of either facility in accordance with its proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The margin of safety associated with the ECCS system is established by acceptance criteria for system performance defined in 10 CFR 50.46. The proposed amendments will not change this acceptance criteria nor the operability requirements for equipment that is used to achieve such performance as demonstrated in the plant safety analyses. Moreover, an integrated assessment of the risk impact of extending the AOT for a single inoperable LPSI train has concluded that the risk contribution is very small, LPSI system reliability can potentially be improved, and the net impact of the proposed change can be risk beneficial. Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

Attorney for licensee: J. R. Newman, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

Florida Power and Light Company, et al., Docket Nos. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: June 21, 1995

Description of amendment request: The proposed amendments will revise the action statements and certain surveillances of TS 3/4.5.1, Safety Injection Tanks (SIT). This proposal is based on the results of a cooperative study performed by participating Combustion Engineering Owners Group members which investigated the impact of a risk-based allowed outage time (AOT) extension, and also included recommendations for line-item TS improvements from NUREG-1366 and Generic Letter 93-05.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The license amendments proposed for St. Lucie Units 1 and 2 incorporate certain line-item Technical Specifications (TS) improvements for the Safety Injection Tanks (SIT), and include an extension of the required action completion/allowed outage time (AOT) from one hour to 72 hours to restore an inoperable SIT (that is still able to perform its safety function) to operable status. In addition, an AOT of 24 hours, based on risk assessment techniques, is proposed for an SIT that may be unable to perform its design function.

The SITs are passive components of the Emergency Core Cooling System (ECCS). As such, they are not accident initiators for any transient evaluated in the plant safety analyses, and an extension of the AOTs for restoring an inoperable SIT to operable status would not increase the probability of occurrence of accidents previously analyzed.

The SITs, in combination with other ECCS components, are used to mitigate the consequences of a loss of coolant accident. The TS revisions will provide a longer AOT for a single inoperable SIT, but do not involve a change to the ECCS configuration or method of operation. The proposed amendments will not change the conditions assumed for the minimum amount of operating equipment needed for accident mitigation. Therefore, the consequences of an accident previously evaluated will not be significantly increased.

In addition to the preceding evaluation, a Probability Safety Assessment (PSA) was performed to quantitatively assess the risk impact of the 24 hour AOT proposal. The impact on the early radiological release probability for design basis events was also evaluated. It was concluded that the risk contribution from this AOT is very small, and that the impact is negligible.

Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendments will not change the physical plant or the modes of plant operation defined in either Facility License. The changes do not involve the addition or modification of equipment, nor do they alter the design of plant systems. Therefore, operation of either facility in accordance with its proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The margin of safety associated with the ECCS system is established by acceptance criteria for system performance defined in 10 CFR 50.46. The proposed amendments will not change this criteria nor the operability requirements for equipment that is used to achieve such performance as demonstrated by the plant safety analyses. Moreover, an integrated assessment of the risk impact of allowing 24 hours to restore an inoperable SIT to operable status has concluded that this impact is very small, and can be offset by averting an unnecessary transition to the shutdown modes. Therefore, operation of either facility in accordance with its proposed amendment would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

Attorney for licensee: J. R. Newman, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

Florida Power and Light Company, et al., Docket Nos. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: August 16, 1995

Description of amendment request: The revisions will modify Technical Specification 3.6.6.1, Shield Building Ventilation System (SBVS), to more effectively address the design functions performed by the SBVS for both the Shield Building (secondary containment) and the Fuel Handling Building.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed license amendment for St. Lucie Unit 2 will clarify the Applicability and the Actions required by Technical Specification (TS) 3.6.6.1, and explicitly account for the dual purpose of the Shield Building Ventilation System (SBVS) to perform design functions for both the Shield Building (secondary containment) and the Fuel Handling Building. The proposed amendment is administrative in nature.

The SBVS only operates when actuated by automatic control signals generated by systems detecting postulated accident conditions. The SBVS is not an accident initiator, the proposed TS changes do not involve any assumptions relative to accident initiators used in the plant safety analyses, and the amendment, therefore, will not impact the probability of occurrence for accidents previously analyzed. Relative to accident consequences, at least one train of the SBVS must operate to fulfill the design function of evacuating filtered air from the Shield Building during the postulated Loss of Coolant Accident; and likewise assumed in the analysis for the Fuel Handling Building during a fuel handling accident. The proposed changes simply remove elements of ambiguity from TS 3.6.6.1; do not reduce the existing operability requirements for the system; and provide further assurance that proper compensatory measures will be taken in the event one or both SBVS trains become inoperable.

Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment is administrative in nature and will not change the physical plant or the modes of plant operation defined in the facility license. The changes do not involve the addition or modification of equipment, nor do they alter the design or methods of operation of plant systems. Plant configurations that are prohibited by TS will not be created by this amendment. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed amendment will not change the SBVS operability requirements nor otherwise alter the basis for any technical specification that is related to the establishment of, or the maintenance of, a nuclear safety margin. The proposed changes are administrative in nature, and are designed to provide assurance that the SBVS capability to perform design functions assumed available in the safety analyses will remain available during the various plant operating modes. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

Attorney for licensee: J. R. Newman, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews

Florida Power and Light Company, et al., Docket Nos. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: August 16, 1995

Description of amendment request: The proposed amendments revise St. Lucie Units 1 and 2 Technical Specifications to relocate selected Technical Specification Monitoring Instrumentation utilizing the Final Policy Statement on Technical Specification Improvement for Nuclear Power Reactors, 58 FR 39132, July 22, 1993. The proposed amendments also include relocation of Technical Specifications related to the Emergency and Security Plan review process utilizing the guidance contained in NRC Generic Letter 93-07, ''Modification of the Technical Specification Administrative Requirements for Emergency and Security Plans.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to the Selected Technical Specification Requirements Related to Instrumentation are administrative in nature in that the specifications for operation and surveillance of the selected Technical Specification instrumentation will be relocated from Appendix A of the facility operating license to the Updated Final Safety Analysis Report (UFSAR) for each unit. Once relocated, future changes will be controlled by 10 CFR 50.59 and the UFSARs updated pursuant to 10 CFR 50.71(e). Relocation of these requirements to the UFSAR is consistent with the NRC ''Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors'' published in the Federal Register (58 FR 39132) dated July 22, 1993.

The selected Technical Specification instruments are not accident initiators nor a part of the success path(s) which function to mitigate accidents evaluated in the plant safety analyses. The proposed Technical Specification change does not involve any change to the configuration or method of operation of any plant equipment that is used to mitigate the consequences of an accident, nor do the changes alter any assumptions or conditions in any of the plant accident analyses. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The Technical Specifications changes associated with Emergency Plan and Security Plan requirements are proposed in accordance with Generic Letter 93 07. The changes being proposed are administrative in nature and do not affect assumptions contained in plant safety analyses, the physical design and/or operation of the plant, nor do they affect Technical Specifications that preserve safety analysis assumptions. Therefore, operation of the facility in accordance with the proposed amendments would not affect the probability or consequences of an accident previously analyzed.

(2) Use of the modified specification would not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed amendment to relocate the existing Technical Specification requirements for selected Technical Specification instrumentation to the UFSAR will not change the physical plant or the modes of plant operation defined in the Facility License. The change does not involve the addition or modification of equipment nor does it alter the design or operation of plant systems. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendments, in accordance with Generic Letter 93-07, change the Technical Specifications to remove the audit of the emergency and security plans and implementing procedures from the list of responsibilities of the Facility Review Group. The changes being proposed are administrative in nature and will not change the physical plant or the modes of operation defined in the Facility License. The change does not involve the addition or modification of equipment nor does it alter the design or operation of plant systems. Therefore, operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Use of the modified specification would not involve a significant reduction in a margin of safety.

The proposed changes are administrative in nature in that operating and surveillance requirements for the selected Technical Specification instrumentation will be relocated from Appendix A of the facility license to the appropriate Updated Final Safety Analysis Report for each unit. These selected instruments are not used to actuate safety-related equipment, provide interlocks, or otherwise perform plant control functions. Conditions evaluated in plant accident and transient analyses do not involve these selected instruments. The proposed changes do not alter the basis for any technical specification that is related to the establishment of, or the maintenance of, a nuclear safety margin. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed amendments, in accordance with Generic Letter 93-07, change the Technical Specifications to remove the audit of the emergency and security plans and implementing procedures from the list of responsibilities of the Facility Review Group. The changes being proposed are administrative in nature and do not alter the bases for assurance that safety-related activities are performed correctly or the basis for any Technical Specification that is related to the establishment of or maintenance of a safety margin. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

Attorney for licensee: J. R. Newman, Morgan, Lewis & Bockius, 1800 M Street, N.W., Washington, DC 20036

NRC Project Director: David B. Matthews

IES Utilities Inc., Docket No. 50-331, Duane Arnold Energy Center, Linn County, Iowa

Date of amendment request: July 21, 1995

Description of amendment request: The proposed amendment would make administrative changes to various sections of the Duane Arnold Energy Center (DAEC) Technical Specifications (TS). These changes replace a conditional surveillance if one emergency service water (ESW) pump or loop is determined to be inoperable (TS 4.8.E.2); credit successful emergency diesel generator (EDG) tests performed in the previous 24 hours (TS 4.8.E.2); clarify the requirements governing spent and new fuel storage in Section 5.5 of the DAEC TS; and eliminate the Operations Committee reviews of procedures in support of the DAEC Emergency Plan and Security Plan, as specified in Sections 6.5 and 6.8 of the TS. DAEC TS Section 4.8.E.2 states the surveillance requirement applicable when one ESW pump or loop is determined to be inoperable. This amendment request deletes the surveillance requirement to physically test the opposite train's EDG and replaces it with a requirement to verify OPERABILITY of the opposite train low pressure core and containment cooling systems and EDG.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed revision does not involve a significant increase in the probability or consequences of an accident previously evaluated. The changes are administrative in nature and are consistent with previously-published NRC guidance. The proposed revision does not change any accident analysis, plant safety analysis or calculations; degrade existing plant programs; or modify any functions of safety related systems or accident mitigation functions for which the DAEC has previously been credited. The proposed revision to the Surveillance Requirements will continue to assure OPERABILITY as required, but eliminate unnecessary operation of an EDG.

2. The proposed revision does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed revision does not alter any plant parameters, revise any safety limit setpoint, or provide any new release pathways. In addition, the proposed revision does not modify the operation or function of any safety-related equipment, nor introduce any new modes of operation, failure modes, or physical changes to the plant.

3. The proposed revision does not involve a significant reduction in a margin of safety. The proposed revision does not alter any plant parameters, revise any safety limit setpoint, or provide any new release pathways. In addition, the proposed revision does not modify the operation or function of any safety-related equipment, nor introduce any new modes of operation, failure modes, or physical changes to the plant.

The NRC staff has reviewed the licensee's analysis and, based on thisreview, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, S.E., Cedar Rapids, Iowa 52401

Attorney for licensee: Jack Newman, Kathleen H. Shea, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036-5869

NRC Project Director: Gail H. Marcus

Nebraska Public Power District, Docket No. 50-298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: May 5, 1995, as revised by letter dated July 14, 1995

Description of amendment request: The proposed changes would amend the Cooper Nuclear Station (CNS) Technical Specifications (TS) sections 3/4.5.F.1, 3.5.F.2, 3.9.B.1, 3.9.B.2, 4.9.A.2, and the associated bases. These changes would revise the TS to: 1) verify that the redundant diesel generator is operable upon the loss of one diesel generator, and implement provisions to verify that the operable diesel generator does not have a common cause failure; 2) incorporate provisions to allow a modified start for the diesel generators; and 3) remove the requirement that the reactor power level be reduced to 25% of rated power upon loss of both diesel generator units or both incoming power sources (start-up and emergency transformers). In addition, the period of time allowed for continued reactor operation with both diesels inoperable would be reduced from 24 to two hours.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

10 CFR 50.91(a)(1), requires that licensee requests for operating license amendments be accompanied by an evaluation of significant hazards posed by the issuance of the amendment. NPPD has reviewed the proposed changes in accordance with 10CFR50.92 and concludes that the changes do not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve a SHC because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed.

Proposed Revision 1:

This proposed revision serves to ensure that an emergency diesel generator is always available to perform on demand and that lowering the number of demands to demonstrate operability reduces the probability of equipment failure. The required action no longer requires the redundant emergency diesel generator to be demonstrated operable immediately. Therefore, this requirement has been deleted from TS 4.5.F.1.

The proposed change includes provisions to determine if the redundant diesel generator has been made inoperable by a common cause failure or perform a demonstration test. The redundant emergency diesel generator will remain in service during the entire period of inoperability of the out of service emergency diesel generator. If a common cause failure cannot be ruled out, the redundant diesel generator will be tested in accordance with the surveillance requirements of TS section 4.9.A.2.a.1 to assure operability.

Since this proposed revision does not affect the design or negatively affect the performance of the diesel generators, the change will not result in an increase in the consequences or probability of an accident previously analyzed. This proposed revision will increase diesel generator reliability and availability, thereby increasing overall plant safety.

Proposed Revision 2:

This proposed revision only affects emergency diesel generator periodic testing. The diesel generators are not accident initiators and the method of testing the diesel generators cannot initiate an accident and therefore will not increase the probability of an accident. This change to the diesel generator testing method does not impact any Updated Safety Analysis Report (USAR) safety analysis. The proposed surveillances will still provide assurance that the diesel generators are available to mitigate the consequences of accidents previously evaluated. Thus the consequences of an accident previously evaluated are not increased.

The revised periodic testing will still demonstrate that the emergency diesel generators are ready to perform their safety function. An overall improvement in diesel engine reliability and availability can be gained by performing diesel generator starts for surveillance testing using engine prelubes, warmups and other manufacturer recommended practices to reduce engine stress and wear. Since this proposed revision does not affect the design or negatively affect the performance of the diesel generators, the change will not result in an increase in the consequences or probability of an accident previously analyzed. This proposed revision will increase diesel generator reliability, thereby increasing overall plant safety.

Proposed Revision 3:

This proposed revision does not affect the operation of the emergency diesel generators or the incoming power sources (start-up and emergency transformers). Both the diesel generators and the incoming power sources function to mitigate the consequences of postulated accidents. As such, removing the requirement to reduce power level upon the loss of both redundant components in either of these systems does not create an increase in the probability of an accident. By eliminating this requirement, the potential for plant transients during power reduction to 25% are also eliminated. Eliminating this requirement will not increase the consequences of a postulated accident because the redundant components will remain available. Additionally, the loss of both offsite power sources condition becomes more restrictive by requiring a plant shutdown instead of notification within 24 hours.

The proposed changes do not alter the conditions or assumptions in any of the Updated Safety Analysis Report (USAR) accident analyses. Since the USAR accident analyses remains bounding, the radiological consequences previously evaluated are not adversely affected by the proposed changes. Therefore, no significant increase in the probability or consequences of an accident previously analyzed would occur.

The proposed rearrangement of information, and rewording of some the TS requirements are included to enhance usability and alleviate any possible confusion. These changes are strictly editorial have no impact, and do not alter technical content or meaning of the specifications. These editorial changes do not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

Proposed Revision 1:

Accidents involving loss of off-site power and single failure have been previously evaluated, and this proposed change does not impact any of those assumptions. This proposed revision does not introduce any new mode of plant operation or new accident precursors, involve any physical alterations to plant configurations, or make changes to system setpoints which could initiate a new or different kind of accident. Operation of the facility in accordance with the proposed revised changes does not create the possibility of a new or different kind of accident from any previously evaluated.

Proposed Revision 2:

This proposed revision only affects emergency diesel generator periodic testing. The diesel generators are not accident initiators and the method of testing the diesel generators cannot initiate an accident. This revision does not relieve the operation of the diesel generator from existing requirements and the diesel generators remain bounded by the assumptions in the USAR accident analysis. The method of testing provides assurance that the diesel generators are available when needed. The proposed revision does not involve any changes in setpoints, plant equipment, plant operation, protective functions, or the design basis of the plant. Therefore, a change in the method of starting the diesel generators during periodic testing would not create a different kind of accident than previously evaluated.

Proposed Revision 3:

This proposed revision does not add or change any equipment or logic, nor do the changes associated with this revision alter any system operability requirements. The proposed changes for this revision do not introduce any new failure modes for any plant system or component important to safety nor has any new limiting failure been identified as a result of the proposed revision. Since there are no changes to the function, or operation of any system, equipment, or component, the possibility of a new or different kind of accident is not created.

The proposed rearrangement of information, and rewording of some [of] the TS requirements are included to enhance usability and alleviate any possible confusion. These changes are strictly editorial have no impact, and do not alter technical content or meaning of the specifications. These editorial changes do not create the possibility of a new or different kind of accident from any previously analyzed.

3. Involve a significant reduction in the margin of safety.

Proposed Revision 1:

This proposed revision does not result in an overall reduction in the margin of safety. The reduction in margin going from ‘‘immediately’’ testing an operable diesel generator to 24 hours to determine no common cause, is offset by the increase in margin resulting from increased diesel generator reliability and availability associated with implementing the vendor recommendations for testing and not exposing the diesel generator to potential grid disturbances when a diesel generator is found to be inoperable. No physical modification to the plant or change in the procedurally prescribed operator actions result from the proposed changes associated with this revision. Operation of the facility in accordance with the proposed revision does not involve a significant reduction in a margin of safety.

Proposed Revision 2:

This proposed revision is made to increase the reliability and availability of the emergency diesel generators thus enhancing the safety of the plant. Changing the way periodic testing of the diesel generators is conducted does not involve a reduction in safety. The test still demonstrates the ability of the diesel generator to start within the time required, and reach rated voltage and frequency as required in the accident analysis. The test also demonstrates the ability of the diesel generator to start reliably, carry the required load, and ensures the capabilities of the cooling system and other support systems are operable. Therefore, assurance that the diesel generators operate within the limits determined to be acceptable continues to be provided. Implementing manufacturer’s recommendations to minimize stress and wear of the diesel engine does not involve a significant reduction in the margin of safety, but rather enhances safety.

Proposed Revision 3:

This proposed revision deletes the requirement to reduce reactor power level to 25% of rated power upon the loss of either both diesel generators or both incoming power sources. The elimination of this requirement will allow the plant to maintain the existing power level rather than subject the plant to an unnecessary transient. Maintaining the plant at the existing power level provides a more stable operating environment. The equipment and components of the diesel generators or the incoming power sources are not impacted in any way as a result of the proposed revisions. The margin of safety for the diesel generators and the incoming power sources are not significantly reduced since these systems are not altered in any way, and will continue to be surveillance tested as required. Assurance of operability is provided by the normal, scheduled surveillances which have been established at a sufficient interval to provide reasonable assurance of operability. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

The proposed rearrangement of information, and rewording of some [of] the TS requirements are included to enhance usability and alleviate any possible confusion. These changes are strictly editorial have no impact, and do not alter technical content or meaning of the specifications. These editorial changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee’s analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. The licensee’s July 14, 1995, letter revised the proposed changes in their letter of May 5, 1995, to further limit the period of time that continued reactor operation would be allowed with both emergency diesel generators inoperable from 24 to two hours. This revision to the proposed changes is more restrictive and does not impact the licensee’s analysis of the criteria of 10 CFR 50.92(c). Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Auburn Public Library, 118 15th Street, Auburn, Nebraska 68305

Attorney for licensee: Mr. John R. McPhail, Nebraska Public Power District, Post Office Box 499, Columbus, Nebraska 68602-0499

NRC Project Director: William D. Beckner

Northeast Nuclear Energy Company (NNECO), Docket No. 50-245, Millstone Nuclear Power Station, Unit 1, New London County, Connecticut

Date of amendment request: August 31, 1995

Description of amendment request: The proposed amendment modifies the definition of HOT SHUTDOWN and COLD SHUTDOWN to specify that the definitions are not applicable during the performance of an inservice hydrostatic and leak test (IHLT). Technical Specification Section 3.6.B and 4.6.B would be modified by adding Section 3.6.B.1.b and 4.6.B.1.b to identify the requirements that must be satisfied to consider the reactor in COLD SHUTDOWN during the performance of an IHLT. In addition, the proposed amendment will change temperature specific requirements on several pages to mode or condition specific requirements; make several editorial changes; and change the associated Bases.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has rovided its analysis of the issue of no significant hazards consideration, which is presented below:

NNECO has reviewed the proposed changes in accordance with 10CFR50.92 and concluded that the changes do not involve a significant hazards consideration (SHC). The bases for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve an SHC because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed.

The proposed changes will allow the reactor to be considered in COLD SHUTDOWN during an IHLT with the average reactor coolant temperature greater than 212°F but less than 280°F. The change to allow the reactor to be in COLD SHUTDOWN during the performance of IHLT will not increase the probability or consequences of an accident. The probability of a leak in the reactor pressure boundary during this testing is not increased by considering the reactor to be in COLD SHUTDOWN. The IHLT is performed near water solid, all control rods inserted, and with an appropriate availability of engineering safety features. The stored energy in the reactor core will be very low and the potential for failed fuel and a subsequent increase in coolant activity are minimal. In addition, secondary containment will be operable and capable of handling airborne radioactivity from leaks that could occur during the performance of an IHLT. Requiring secondary containment to be operable will further ensure that potential airborne radiation from leaks will be filtered by one or both trains of SBGT [standby gas treatment], thereby limiting releases to the environment. Therefore, the changes will not significantly increase the consequences of an accident.

In the unlikely event of a large pressure boundary leak, the reactor vessel would rapidly depressurize, allowing one or both of the operable core spray systems to operate. Small system leaks would be detected by leakage inspections before significant inventory loss occurred, since leakage inspections are an integral part of the IHLT program.

Based upon the above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

The IHLT conditions remain unchanged. The potential for a system leak remains unchanged since the reactor coolant system is designed for temperatures exceeding 500°F with similar pressures. The change in operable engineered safety features available to mitigate a postulated accident does not reduce the ability to safely mitigate a postulated accident. Adequate ECCS [emergency core cooling system] equipment will be available to mitigate a LOCA [loss of coolant accident] with an assumed single failure. Therefore, this will not create the possibility of a new or different kind of accident from any previously evaluated.

3. Will not involve a significant reduction in a margin of safety.

The proposed changes will not have any significant impact on any design basis accident or safety limit. The various engineered safety features which are required by the proposed change will ensure appropriate mitigation of postulated events. Since the test is performed at a near water solid condition and at low decay heat values, no fuel damage is expected in case of an accident such as a LOCA. Nevertheless, secondary containment and the SBGT system will be maintained operable to process air-borne radioactivity from a steam leak that could occur during the performance of the IHLT. Therefore, the proposed change does not constitute a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270.

NRC Project Director: Phillip F. McKee

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London, Connecticut

Date of amendment request: August 31, 1995

Description of amendment request: The proposed change to the Millstone 2 Technical Specifications would remove the phrase ''other than Millstone Unit No. 2'' from Section 6.3.1 on page 6-2.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

NNECO has reviewed the proposed change in accordance with 10CFR50.92 and concluded that the change does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed change does not involve an SHC because the change would not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed.

The proposed change does not affect any system or equipment of Millstone Unit No. 2. The proposed change does not affect the qualification of any of the licensed individuals involved in the day-to-day operation of Millstone Unit No. 2. The proposed change corrects a statement which could be interpreted such that an individual who once held a Millstone Unit No. 2 SRO [Senior Reactor Operator] license would not be eligible to be Operations Manager. Since this change does not affect any equipment or operating procedures, does not affect the level of expertise and training required for on-shift personnel, and does not reduce the level of expertise required of operations management, this change does not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

This change does not affect any equipment or operating procedures, does not affect the level of expertise and training required for on-shift personnel, and does not reduce the level of expertise required of operations management. Therefore, this change does not create the possibility of a new or different kind of accident.

3. Involve a significant reduction in the margin of safety.

This change eliminates a phrase which could be interpreted to prevent an individual who had possessed a Millstone Unit No. 2 SRO license from becoming the Operations Manager. The training and experience necessary to possess a Millstone Unit No. 2 SRO license is equivalent to that of other PWRs. Therefore, this proposed change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, CT 06141-0270.

NRC Project Director: Phillip F. McKee

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: July 28, 1995

Description of amendment request: The proposed amendment would eliminate the Technical Specifications requirements to perform 10 CFR 50, Appendix J, Type C hydrostatic testing on certain valves that are within closed systems and are assured a water seal following a Design Basis Accident.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed TS changes do not involve a significant increase in the probability or *consequences of an accident previously evaluated.*

The primary containment (drywell and suppression pool) and the affected closed systems are accident mitigators not accident initiators. The proposed change to the scope of Appendix J, Type C testing does not affect the probability of the DBA [Design Basis Accident]. The valves will continue to be maintained in an operable state, and in their current design configuration. There is no correlation between the scope of Appendix J, Type C testing and accident probability. There are no physical or operational changes to the containment structure, system or components being made as a result of the proposed changes. Therefore, the consequences of a malfunction of equipment important to safety is not increased from those previously evaluated.

The consequences of loss-of-coolant accidents (LOCAs) under the proposed change were considered where a single active failure of a containment isolation valve (CIV) or a passive failure of the closed system were reviewed, within the limits of the existing licensing basis. Under the existing licensing basis, a pipe rupture of the seismically qualified ECCS piping does not have to be assumed concurrent with the LOCA, except if it is a consequence of the LOCA. Consideration of consequential failures can be eliminated, since a LOCA inside containment is separated from the affected piping by the containment structure. Consideration of consequential failures of the ECCS piping from LOCAs outside containment are outside the Appendix J design considerations. A single active failure of the CIV, under the LOCA condition, can be accommodated since the closed and water sealed system piping remains as the leakage barrier. The ECCS passive failure criterion does require consideration of system leaks, but not pipe breaks, beyond the initiating LOCA. The capability to make-up water inventory to the suppression pool is adequate to ensure that postulated seat leakage and pipe leakage does not result in a condition that jeopardizes pool level. Make-up capability exists for the suppression pool via the Condensate Storage Tank and Ultimate Heat Sink Spray Pond. Operator actions to make-up the suppression pool are delineated in existing Operating Procedures.

The subject valves are single isolation valves associated with lines that penetrate the primary containment, but are not connected directly to the primary containment atmosphere or the reactor coolant pressure boundary. This configuration is described in the LGS UFSAR, Section 6.2.4.3.1.3.1, which states ‘‘the systems which the lines from the suppression pool connect to outside containment are closed systems meeting the appropriate requirements of closed systems.’’ The integrity of these closed systems are also monitored and controlled in accordance with TS Section 6.8.4.a. Any leakage that may escape the confines of the closed system will be contained within the Reactor Building, treated by standby gas and radwaste systems, and, therefore, are within the existing LGS licensing bases.

Finally, the affected penetrations will continue to be subjected to the periodic 10 CFR 50, Appendix J, Type A test (Integrated Containment Leakage Rate Test).

The suppression pool level is designed and operated so that water level is maintained in accordance with current TS, and the associated bases. The supply of water in the suppression pool is assured for 30 days during all DBA, post-accident modes of operation. The lowest water level which the suppression pool will reach was analyzed, and it was determined that the affected lines will remain below this minimum level, thereby assuring a water seal. The valves will continue to be tested and maintained to ensure their operability, and the closed systems’ integrity will continue to be monitored and controlled in accordance with TS 6.8.4.a and the performance of the periodic 10 CFR 50, Appendix J, Type A test.

Therefore, the proposed changes will not increase the probability or consequences of an accident previously evaluated.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not change the plant response to accident scenarios, and do not introduce new or different scenarios. The primary containment (drywell and suppression pool) and the affected closed systems are accident mitigators not accident initiators. The proposed change to the scope of Appendix J, Type C hydrostatic testing maintains the existing barriers to primary containment bypass leakage by the assurance that a water seal will be maintained for 30 days during all DBA, post-accident modes of operation. The valves will continue to be tested and maintained to ensure their operability, and the closed systems’ integrity will continue to be monitored and controlled in accordance with TS 6.8.4.a. Therefore, the proposed changes cannot cause an accident, and the plant response to the design basis events is unchanged, whereby the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed TS changes do not involve a significant reduction in a margin of safety.

The water seal provided by the assurance of a minimum suppression pool level will prevent post-accident containment bypass leakage. Appendix J does not require air leak testing of the valves since the 30 day post-accident supply of water is maintained. In addition, the closed systems’ integrity is monitored and controlled in accordance with TS 6.8.4.a. Any leakage that may escape the confines of the closed system will be contained within the Reactor Building, and is within the existing LGS licensing bases. Therefore, the proposed TS changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee’s analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: John F. Stolz

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: July 28, 1995

Description of amendment request: The proposed amendments, which are consistent with the Improved Standard Technical Specifications (NUREG-1433), delete the operability and surveillance requirements involving secondary containment differential pressure instrumentation.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed Technical Specifications (TS) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

Deleting the operability and surveillance requirements for the secondary containment differential pressure instrumentation does not involve any changes to the design, function, or operation of any plant components or safety-related systems. There are no changes to the separation, redundancy, qualification, quality assurance or fire protection requirements for the associated components and systems, nor are there any new failure modes created. This activity only removes operability and surveillance requirements from the Technical Specifications for selected plant components associated with the secondary containment differential pressure trip functions. No credit for operation of these trip functions is taken in any design basis accidents valuated in the SAR [Safety Analysis Report]. Thesecomponents will be maintained in accordance with the plant preventive maintenance program. The failure of any of these components does not result in the occurrence of an accident. Consequently, there is no increase in the probability of occurrence of an accident previously evaluated in the SAR.

The Outside Atmosphere to Reactor Enclosure Delta Pressure-Low and Outside Atmosphere To Refueling Area Delta Pressure-Low trip functions are not symptomatic of a design basis accident. No credit for operation of the trip functions is taken in any design basis accidents evaluated in the SAR. Neither failure of the differential pressure components nor failure to generate the associated trip functions affects the consequences of an accident previously evaluated in the SAR. The appropriate accident prevention and mitigation actions are generated from other plant parameters symptomatic of an accident. Sufficient plant parameters symptomatic of a design basis accident are monitored to initiate the appropriate actions as evaluated in the SAR. Furthermore, all safety-related systems will still be able to perform all of their design basis safety-related functions. Consequently, there is no increase in the consequences of an accident previously evaluated in the SAR.

Therefore, the proposed TS changes do not involve an increase in the probability or consequences of an accident previously evaluated.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The failure of the differential pressure automatic isolation instrumentation components does not result in the occurrence of an accident. The failure to generate the associated trip functions does not result in the occurrence of an accident. This activity does not involve any changes to the design, function, or operation of any plant components or safety-related systems. There are no changes to the separation, redundancy, qualification, quality assurance or fire protection requirements for the associated components and systems. These components will be maintained in accordance with the plant preventative maintenance program. Consequently, there is no possibility of an accident of a different type than previously evaluated in the SAR.

Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed TS changes do not involve a significant reduction in a margin of safety.

The ability of secondary containment to minimize any ground level release of radioactive material which may result from any accident is not affected. Surveillance and operability requirements for secondary containment SGTS [Standby Gas Treatment System] and RERS [Reactor Enclosure Recirculation System] are not changed by this activity. Draw down time, leakage factors, secondary containment system ratings, and secondary containment system response to a LOCA [Loss of Coolant Accident] or refueling accident are not affected by this activity. SGTS and RERS initiation will continue to occur when plant parameters symptomatic of a LOCA or refueling accident exceed predetermined values. There are no changes to the inputs for the post-LOCA offsite dose analysis.

Therefore, the proposed TS changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: John F. Stolz

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: July 28, 1995

Description of amendment request: The proposed amendment would modify Technical Specifications (TS) Surveillance Requirements 4.9.1.1, 4.9.1.2, 4.9.3, 4.9.5, and 4.9.8 to delete specific requirements to perform surveillances just prior to beginning or resuming core alterations or control rod withdrawal associated with refueling activities. This proposed TS change would delete the phrase ''incore instrumentation'' from the footnote in TS Section 3/4.9.5, ''Communications.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed Technical Specifications (TS) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS changes do not involve any physical changes to plant systems or equipment. The proposed TS changes only delete those Surveillance Requirements (SRs) pertaining to the performance of tests just prior to beginning or resuming core alterations or control rod withdrawal, and revises a footnote description to be consistent with the current TS definition of ''Core Alteration.'' The proposed TS changes do not revise any of the other applicable periodic SRs, or modify any procedural controls currently in place governing fuel handling operations. The periodic surveillance test frequencies provide adequate assurance that the equipment will remain in an operable condition. The normal periodic surveillance intervals bound those surveillance intervals for the tests that are being altered by this proposed TS change. In the event that one of the periodic surveillances has not been performed within the specified time interval, entry into the specified condition (i.e., performance of core alterations, control rod withdrawal, or handling of fuel or control rods) is not permitted as required by TS 4.0.4 until the surveillance has been satisfactorily completed.

The consequences of an accident are not increased by the proposed TS changes, since the changes only involve revising the frequency of conducting surveillance tests. The method of operation or performance of plant structures, systems, or plant components are not affected by the proposed TS changes. The proposed TS changes will not impact the operation of any fuel handling equipment, and therefore, the potential for a Fuel Handling Accident as described in Section 15.7.4 of the LGS [Limerick Generating Station] Updated Final Safety Analysis Report (UFSAR) is not increased.

In addition, any unexpected reduction of water level in the reactor cavity or fuel pool at the start of fuel handling or control rod handling will be immediately apparent to operators by direct observation. Plant procedures utilized by the refueling personnel require the suspension of core component transfers in the event of loss of water inventory.

Therefore, the proposed TS change does not involve an increase in the probability or consequences of an accident previously evaluated.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed TS changes only involve changes to the frequency in which the specified surveillance tests are performed. The proposed TS changes do not revise any of the other applicable periodic SRs, or modify any procedural controls currently in place governing fuel handling operations. The periodic surveillance test frequencies provide adequate assurance that the equipment will remain in operable condition. The periodic surveillance intervals bound those surveillance intervals for the tests that are being altered by this proposed TS change. The refueling interlock system combined with strict procedural controls provide multiple barriers to preclude an inadvertent criticality.

The proposed TS changes do not involve any physical changes to plant systems or equipment. The proposed TS changes do not alter the configuration of the plant or the way that the plant is operated. The associated plant equipment will continue to function as designed. This equipment is not designed to perform any other function than it is presently capable of, and therefore, will not affect the operation of any other plant equipment.

Therefore, the proposed TS change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed TS changes do not involve a significant reduction in a margin of safety.

The proposed TS changes do not involve any physical changes to plant systems or equipment. The reactor will continue to be maintained subcritical during refueling operations and reactor water level will be maintained at the required level (i.e., above the vessel flange). The proposed TS changes do not affect the operation of other plant systems and equipment essential in maintaining reactor water temperature during refueling operations, or the capability in responding to a postulated Fuel Handling Accident.

The proposed changes do not adversely affect reliability of the refueling interlocks or refuel platform communications equipment.

Since the proposed changes only impact the frequency in which certain surveillance tests are performed, and do not change the plant configuration or setpoints, there is substantial assurance that the reactor will be maintained subcritical during refueling.

Therefore, the proposed TS changes do not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: John F. Stolz

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: July 28, 1995

Description of amendment request: The proposed amendment would revise Technical Specifications Table 4.3.1.1-1, ''Reactor Protection System Instrumentation Surveillance Requirements'', to reflect changes to the surveillance test frequency requirements for various Reactor Protection System [RPS] instrumentation.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed Technical Specifications (TS) changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

In all of the applicable SAR [Safety Analysis Report] evaluated events, the IRM [Intermediate Range Monitor] and APRM [Average Range Power Monitor] instrumentation is credited for performing a mitigating function (i.e., initiating a scram), to terminate the transient prior to a safety limit being exceeded. The proposed TS changes do not alter the RPS configuration, or RPS instrumentation setpoints, nor do they change the manner in which the IRM and APRM instrumentation carry out the scram functions. Therefore the consequences of any potential malfunction of equipment important to safety will remain unchanged.

In each case where a startup surveillance test requirement is proposed to be deleted, (i.e., IRM and APRM), the normal surveillance test frequency specified for the required Operational Condition remains unchanged (except for the APRM Upscale Setdown functional test). The startup surveillance requirement is conservatively bounded by the normal surveillance test interval which is greater than or equal to any interval associated with the startup surveillance requirement and ensures that the IRM and APRM instrumentation reliability is unchanged. This is in accordance with the Improved Standard Technical Specifications, NUREG-1433, issued September 28, 1992.

The reliability of the APRM Upscale Setdown scram function will not be decreased due to changing the functional test frequency from Weekly (W), to Quarterly (Q), in Operational Conditions 2, 3, and 5 (Startup, Hot Shutdown and Refueling, respectively). Plant operational data taken from each of the APRM calibration/functional tests performed since August 1992 until present at LGS Units 1 and 2, shows that setpoint reliability will be maintained if the functional test frequency is increased to quarterly, as proposed. Presently, each time an APRM calibration/functional test is performed, both the Upscale Setdown and the Flow Reference scram circuits are tested. The results of the quarterly tests confirm that the APRM Upscale Setdown function already has over 2.5 years of performance without failure in Operational Condition 1, thus being extremely reliable.

Therefore, the proposed TS changes do not involve an increase in the probability or consequences of an accident previously evaluated.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed TS changes affect only the required surveillance test intervals, not the RPS configuration or RPS instrumentation setpoints. The proposed TS changes do not introduce a new failure mode for the IRM or APRM instrumentation. Plant operating experience data confirms that at LGS Units 1 and 2, the IRM and APRM instrumentation will continue to perform their safety function as currently designed, with the same degree of reliability.

The proposed TS changes do not alter the configuration of the plant, nor the way the plant is operated.

Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident, from any accident previously evaluated.

3. The proposed TS changes do not involve a significant reduction in a margin of safety.

The following TS Bases were reviewed for potential reduction in the margin of safety:

B 2.2.1 Reactor Protection System Instrumentation Setpoints

B 3/4.1.4 Control Rod Program Controls

B 3/4.2 Power Distribution Limits

B 3/4.3.1 Reactor Protection System Instrumentation

B 3/4.3.6 Control Rod Block Instrumentation

The surveillance test frequency changes proposed for the RPS instrumentation section of TS do not adversely affect the IRM or APRM instrumentation, which will continue to perform the RPS functions required to maintain the present margin of safety. Changes to the IRM instrumentation startup surveillance intervals are already bounded by the existing surveillance requirements, and are in accordance with the Improved Standard Technical Specifications, NUREG-1433, issued September 28, 1992. The same statement applies to the APRM instrumentation, with respect to deletion of the startup surveillance requirement. The change of the APRM Upscale Setdown Channel functional test surveillance interval from Weekly to Quarterly was evaluated to ensure that the APRM instrumentation would perform that function, with the same degree of reliability as presently experienced. A review of the plant operating experience data at LGS Units 1 and 2 shows that APRM instrumentation is extremely reliable for a quarterly surveillance test interval. The proposed TS changes do not modify plant configuration, RPS instrumentation setpoints, or RPS operation. The margin of safety remains unchanged.

Therefore, the proposed TS changes do not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: John F. Stolz

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: August 1, 1995

Description of amendment request: The proposed amendment would modify Technical Specifications Section 3/4.9.1, ''Reactor Mode Switch,'' in order to provide alternate actions to allow the continuation of core alterations in the event certain Reactor Manual Control System (RMCS) and refueling interlocks are inoperable, while preserving the intended function of the inoperable interlocks.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed Technical Specifications (TS) changes do not involve a significant

increase in the probability or consequences of an accident previously evaluated.

The refueling and one-rod-out interlocks impose barriers to preclude an inadvertent criticality during refueling operations. Section 7.7.2.15.1 of the LGS Updated Final Safety Analysis Report (UFSAR) clearly delineates the functions of the interlocks and the criteria used in assessing correct refueling and one-rod-out interlock operation in the following statement.

In all cases, correct operation of the refueling interlock prevents either the operation of loaded refueling equipment over the core when any control rod is withdrawn, or the withdrawal of any control rod when fuel-loaded refueling equipment is operating over the core. In addition, when the reactor mode switch is in REFUEL position, only one rod can be withdrawn, and selection of a second control rod initiates a rod block.

The proposed TS changes provide operational flexibility while strictly conforming to, and preserving, the intended function of the refueling and one-rod-out interlocks. The proposed TS changes that could affect interlock capabilities are identified below, along with the appropriate justification to substantiate that the proposed TS changes will not result in an increase in the probability or consequences of an accident previously evaluated.

a.TS Section 3.9.1, ACTION Statement b. The proposed change to this existing TS ACTION will add a verification that all control rods are fully inserted, and then disabled from being withdrawn as a suitable alternative to placing the reactor mode switch in the SHUTDOWN position when the one-rod-out interlock is not operable. In addition, the proposed change to this TS section includes a caveat of non-applicability for those control rods already removed in accordance with requirements stipulated in TS Sections 3.9.10.1 and 3.9.10.2. As indicated in LGS UFSAR which is described in the statement above, it is expected that the refuel and one-rod-out interlocks will permit the withdrawal of only one (1) control rod at a time with the reactor mode switch in the REFUEL position, and no control rods can be moved when fuel-loaded refueling equipment is operating over the core. By verifying all control rods are inserted, then disabling withdraw capabilities of all rods, as requested, the most limiting requirements for control rod motion will be met. The potential for having more than one (1) control rod out at a time, or having any control rod not fully inserted while fuel-loaded refueling equipment is operating over the core, does not exist when applying the alternative. Therefore, the intended functions of the refuel and one-rod-out interlocks are operationally preserved. Since TS Sections 3.9.10.1 and 3.9.10.2 have specific requirements for removing surrounding fuel prior to control rod blade removal, the control rods already removed are no longer required to carry out a safety function in the defueled cell, and as a result would not apply for this specific proposed TS change. From a control rod withdrawal perspective, there is no functional difference between the proposed TS change and the existing, and still remaining, TS ACTION of locking the reactor mode switch in SHUTDOWN position.

b. TS Section 3.9.1, ACTION Statement c. This existing TS ACTION requires that core alterations be suspended in the event that a refueling interlock is not operable. The proposed TS change to this TS ACTION leaves this requirement in place, but makes this ACTION specifically applicable to the refueling platform, and adds three (3) new additional ACTION alternatives. The wording for changes to this TS section are such that implementation of any one of the three (3) new alternatives can be substituted for suspending core alterations. The proposed wording for these three (3) new alternatives and justification is provided below.

1) Verify all control rods are fully inserted and disable withdraw capabilities of all control rods***.

Since this alternative ensures all control rods are, and will remain fully inserted, all required conditions of the associated refueling and one-rod-out interlocks are met. The refueling interlock is satisfied since a fuel-loaded refueling platform operating over the core would be assured that all control rods are fully inserted and prevented from being withdrawn. The one-rod-out interlock is satisfied since control rod withdrawal is disabled for all control rods, which is an even more conservative requirement than the one-rod-out interlock itself. While operating in this configuration, there will be no associated travel or hoist restrictions for the refueling platform over the core, which is normal for the current refuel interlock design. The potential for having any control rod not fully inserted while a fuel-loaded refueling platform is operating over the core, does not exist when applying this proposed alternative. Therefore, the intended function of the refueling platform refuel interlocks are operationally preserved with the implementation of this proposed alternative, and there will be no increase in the probability of occurrence of an accident. This proposed alternative also maintains an exclusion (via a reference to the proposed *** footnote) for control rods removed in accordance [with] TS Sections 3.9.10.1 and 3.9.10.2. This exclusion does not apply to inadvertent criticality concerns, as previously discussed in Item 1.a above.

2) Verify Refuel Platform is not over core (limit switches not reached) and disable refuel platform travel over core.

As previously stated above, LGS UFSAR Section 7.7.2.15.1 stipulates that the refueling platform position interlocks initiate a control rod block whenever a fuel-loaded refueling platform is over the core, and stop a fuel-loaded refueling platform from moving over the core if a control rod is already withdrawn. This specific proposed TS change satisfies both these requirements by precluding the possibility of the platform from being over the core. If a control rod is being withdrawn, the platform will not be over the core, and the withdrawal will be in accordance with the current design. If a control rod is already withdrawn, disabling platform travel over the core, before reaching the over-core limit switches, is performing the same function as the existing refueling platform reverse and forward motion blocks. Therefore, the potential for having any control rod not fully inserted while a fuel-loaded refueling platform is operating over the core, does not exist when applying this proposed alternative. The intended refueling interlock functions are operationally preserved with the implementation of this proposed alternative.

3) Verify that no Refuel Platform hoist is loaded and disable all Refuel Platform hoists from picking up (grappling) a load.

As previously stated above, UFSAR Section 7.7.2.15.1 stipulates that blocking control rod withdrawal with a refueling platform over the core, and restricting refueling platform travel from going over the core with a control rod already withdrawn, are based on the refueling platform hoist being fuel-loaded. An unloaded platform without grappling capabilities poses no threat to erroneous fuel bundle or control rod removal, and eliminates the potential for having any control rod not fully inserted while a fuel-loaded refueling platform is operating over the core. Therefore, implementing this proposed alternative operationally preserves the intended interlock functions.

c. TS Section 3.9.1, ACTION Statement d. The proposed TS change adds this new TS ACTION section to specify the refueling interlock requirements for the service platform, since the applicability of ACTION Statement c above is being revised to specifically address refueling interlocks associated with the refueling platform. The proposed TS changes for new this TS section retain the existing requirement to suspend core alterations if the service platform associated refueling interlock is not operable, unless the service platform is not installed over vessel. The specific proposed TS changes add two (2) new additional ACTION alternatives. The proposed wording for these two (2) new ACTION statements are such that implementation of any one of the two (2) new alternatives can be substituted for suspending core alterations. Not enforcing operability requirements on the service platform refueling interlocks when the service platform is not over the vessel does not pose an inadvertent criticality concern since there is no associated hoist to manipulate fuel bundles or control rods. These two (2) new alternatives are not applicable unless the service platform is installed over the vessel, and are described below.

1) Verify all control rods are fully inserted and disable withdraw capabilities of all control rods***.

This alternative ensures that all control rods are, and will remain, fully inserted which meets the required conditions for proper refueling and one-rod-out interlock operation. The refueling interlock is satisfied since a fuel-loaded service platform hoist operating over-core is assured that all control rods are fully inserted and prevented from being withdrawn. The one-rod-out interlock is satisfied since all control rods are disabled, an even more conservative requirement than the one-rod-out interlock itself. While operating in this configuration, there will be no associated hoist restrictions for the service platform, which is normal for the current refuel interlock design. The potential for having any control rod not fully inserted while a service platform hoist is fuel-loaded over the core, does not exist when utilizing this proposed alternative. Therefore, the intended function of the service platform refuel interlocks are operationally preserved with the implementation of this proposed alternative. This proposed alternative also maintains an exclusion (via a reference to the proposed *** footnote) for control rods removed in accordance with the requirements of TS Sections 3.9.10.1 and 3.9.10.2. This exclusion is not applicable to inadvertent criticality concerns as discussed in Item 1.a above.

2) Verify Service Platform hoist is not loaded and disable Service Platform hoist from picking up (grappling) a load.

As previously described above, UFSAR Section 7.7.2.15.1 stipulates that blocking control rod withdrawal with the service platform over the core is based on the service platform hoist being fuel-loaded. An unloaded hoist without grappling capabilities poses no threat to erroneous fuel bundle or control rod removal and eliminates the potential for having any control rod not fully inserted while a fuel-loaded service platform is operating over the core. Therefore, implementing this proposed alternative operationally preserves the intended refueling interlock functions.

As discussed in the LGS UFSAR, the use of the refueling and one-rod-out interlocks are evaluated from a prevention, not a mitigation, perspective. A Rod Withdrawal Error (RWE) transient event during refueling is concerned with an inadvertent criticality, and assumes the reactor vessel head is off, and the plant is shutdown (i.e, Operating State A). As described in the LGS UFSAR under Nuclear Safety Operational Analysis (NSOA) Event 16, it is assumed that the Reactor Protection System (RPS) terminates the event should the reactor actually reach Operating State B (i.e., head off and not shut down), which is conditional on the reactor mode switch being in the STARTUP position. The proposed TS changes only pertain to the refueling and one-rod-out interlocks. Since these interlocks act only in a preventive mode, the consequences of an inadvertent criticality accident during refueling remain unchanged.

Since the proposed TS changes are limited to the one-rod-out and refueling interlocks, they do not affect the reliability of the associated equipment. The proposed TS changes specify alternative actions that can be taken in the event that an interlock is inoperable. These alternative actions serve to ensure the failed interlock function is preserved, and do not affect the probability of malfunction of the interlocks.

The one-rod-out and refueling interlocks, as evaluated in the LGS UFSAR, are designed to preclude an inadvertent criticality during refueling operations by placing strict controls on fuel bundle and control rod manipulations, using the following methods.

a. Preventing operation of a fuel-loaded refueling platform or service platform hoist while over the core if a control rod is already withdrawn.

b. Preventing a fuel-loaded refueling platform from traveling over the core if a control rod is already withdrawn.

c. Preventing any control rod from being withdrawn if a fuel-loaded refueling platform or service platform is already operating over the core.

d. Preventing the withdrawal of more than one control rod at a time with the reactor mode switch in the REFUEL position.

The LGS UFSAR indicates that a single component failure does not cause an interlock failure and that a single interlock failure does not cause an accident. The proposed TS changes provide alternative actions that can be taken in the event of an associated component or interlock malfunction. Implementing the proposed TS changes will continue to ensure that the intended interlock functions are maintained and operationally preserved, as described in the LGS UFSAR.

Therefore, the proposed TS changes do not involve an increase in the probability or consequences of an accident previously evaluated.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed TS changes only pertain to the refueling and one-rod-out interlocks. The refueling and one-rod-out interlocks impose barriers to preclude an inadvertent criticality during refueling operations. The proposed TS changes provide operational flexibility, while strictly conforming to, and preserving, the intended function of the refueling and one-rod-out interlocks. There is no other potential failure mode for these interlocks than has already been evaluated and described in the LGS UFSAR. Implementation of these proposed changes will maintain and operationally preserve the intended interlock functions. Therefore, the malfunction of any associated component or interlock will not adversely impact the plant and any other equipment important to safety, directly or indirectly.

Therefore, the proposed TS changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed TS changes do not involve a significant reduction in a margin of safety.

The proposed TS changes only affect the TS associated with the one-rod-out and refueling interlocks. The associated TS Bases Section 3/4.9, ''Refueling Operations,'' states that the one-rod-out and refueling interlocks maintain conditions during refueling activities that reinforce refueling procedures and reduce the potential for the probability of occurrence of each of the following conditions:

a. Inadvertent criticality,

b. Damage to reactor internals or fuel assemblies, and

c. Exposure of personnel to excessive radioactivity.

The proposed TS changes do not adversely affect the one-rod-out or refueling interlocks. The associated interlocks will continue to perform the refueling functions required to maintain the present margin of safety. The proposed TS changes only contain alternative actions that can be taken in the event an interlock is inoperable. These proposed alternative actions ensure that the intent of the interlocks is preserved, and that there is no reduction in the ability of the interlocks to maintain adequate refueling conditions.

The proposed TS changes will preserve the intended interlock functions, and maintain the existing level of protection against refueling errors that could lead to an inadvertent criticality, damage to reactor internals or fuel assemblies, or excessive personnel radiation exposure. The one-rod-out and refueling interlocks will continue to function with their present degree of reliability. The proposed TS changes will continue to maintain strict controls on fuel bundle and control rod manipulations to avoid inadvertent criticality. The proposed TS changes provide the same level of assurance regar[d]ing the manipulation of control rods during refueling operations as that currently described in the LGS UFSAR, and as discussed below.

a. Preventing operation of a fuel-loaded refueling platform or service platform hoist while over the core if a control rod is already withdrawn.

b. Preventing a fuel-loaded refueling platform from traveling over the core if a control rod is already withdrawn.

c. Preventing any control rod from being withdrawn if fuel-loaded refueling platform or service platform is already operating over the core.

d. Preventing the withdrawal of more than one control rod at a time with the reactor mode switch in the REFUEL position.

Therefore, the proposed TS changes do not involve a reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this eview, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: John F. Stolz

South Carolina Electric & Gas Company (SCE&G), South Carolina Public Service Authority, Docket No. 50-395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of amendment request: February 21, 1995, as revised on August 31, 1995

Description of amendment request: The proposed amendment would revise the Technical Specifications (TS) to reflect changes to 10 CFR Part 20 (including Appendix B, Table 2 concentrations) and provide additional administrative corrections.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The probability or consequences of an accident previously evaluated does not involve a significant increase.

The proposed TS changes showing the relocation of the old 10 CFR 20.106 requirements to the new 10 CFR 20.1302, the old 10 CFR 20.203(c)(2) requirements to the new 10 CFR 20.1601(a), and the old 10 CFR 20.407 requirements to the new 10 CFR 20.2206(b) will not involve a significant increase in the probability or consequences of an accident previously evaluated because there will be no change in the types and amounts of effluents that will be released, nor will there be an increase in individual or cumulative occupational radiation exposures.

The proposed revision to the liquid and gaseous release rate limits will not involve a significant increase in the probability or consequences of an accident previously evaluated because there will be no change in the types and amounts of effluents that will be released, nor will there be an increase in individual or cumulative occupational radiation exposures. This is only a change to the method of (algorithm) determining release rate limits and will not change net limits or change the more restrictive 10 CFR 50 Appendix I dose limits.

The proposed revision to the radioactive material quantity in the settling pond and its associated TS Bases will not involve a significant increase in the probability or consequences of an accident previously evaluated because there will be no change in the types of effluents that will be released, nor will there be an increase in individual or cumulative occupational radiation exposures. This is only a change to the quantity of radioactive material in the settling pond and will conservatively lower net limits.

The proposed revision to the TS bases for the liquid holdup tank activity limit will not involve a significant increase in the probability or consequences of an accident previously evaluated because there will be no change in the types and amounts of effluents that will be released, nor will there be an increase in individual or cumulative occupational radiation exposures. The curie limit is not affected, therefore, the change does not represent a decrease in the level of control previously evaluated.

The proposed revision to the distance at which dose rates are measured from the radiation source or surface will not involve a significant increase in the probability or consequences of an accident previously evaluated because there will be no increase in the individual or cumulative occupational radiation exposures. The change in distance is conservative in its effect on worker protection and is in conformance with 10 CFR 20.1601 requirements.

2. The possibility of a new or different kind of accident from any previously evaluated is not created.

The proposed TS changes showing the relocation of the old 10 CFR 20.106 requirements to the new 10 CFR 20.1302, relocation of the old 10 CFR 20.203(c)(2) requirements to the new 10 CFR 20.1601(a), and relocation of the old 10 CFR 20.407 requirements to the new 10 CFR 20.2206(b) will not create the possibility of a new or different kind of accident from any previously evaluated because the revisions are administrative and will not change the types and amounts of effluents that will be released.

The proposed revision to the liquid and gaseous release rate limits will not create the possibility of a new or different kind of accident from any previously evaluated because the revision is administrative and will not change the types and amounts of effluents that will be released.

The proposed revision to the quantity of radioactive material in the settling pond and its associated TS Bases will not create the possibility of a new or different kind of accident from any previously evaluated because the revision will not change the types of effluents that will be released. This is only a change to the quantity of radioactive material in the settling pond and will conservatively lower net limits.

The proposed revision to the TS bases for the liquid holdup tank activity limit will not create the possibility of a new or different kind of accident from any previously evaluated because the revision is administrative and will not change the types and amounts of effluents that will be released.

Implementation of the more conservative distance at which dose rates are measured will not create the possibility of a new or different kind of accident from any previously evaluated.

3. A significant reduction in a margin of safety is not involved.

The proposed revisions due to the location of requirements will not reduce a margin of safety because they are administrative in nature. No equipment or procedural changes are postulated. There is no impact on any margin of safety.

The proposed revision to liquid and gaseous release rate limits will not reduce a margin of safety because it is administrative in nature. These revisions preserve the existing level of effluent control. No changes to the more restrictive 10 CFR 50 Appendix I dose limits are made. There are no equipment or operational procedure changes, therefore, no accidents of any kind will be created by this change.

The proposed revision to the quantity of radioactive material in the settling pond and its associated TS Bases will not reduce a margin of safety because it is conservative in nature and preserves the existing level of effluent control. There are no equipment or operational procedure changes required, therefore, no accidents of any kind will be created by this change.

The proposed revision to the TS bases for the liquid holdup tank activity limit will not reduce a margin of safety because it is administrative in nature and preserve[s] the existing level of effluent control. No equipment or procedural changes are postulated. There is no impact on any margin of safety.

The change in distance for a High Radiation Area classification from 18 in.(45 cm) to (30 cm)12 in. from the radiation source or surface will not reduce the margin of safety because this change will reduce the worker's stay time in the area and therefore minimize exposure.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180

Attorney for licensee: Randolph R. Mahan, South Carolina Electric & Gas Company, Post Office Box 764, Columbia, South Carolina 29218

NRC Project Director: Frederick J. Hebdon

Southern California Edison Company, et al., Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of amendment requests: July 19, 1995

Description of amendment requests: The licensee proposes to revise technical specifications (TSs) to (1) support modifications to the containment area radiation monitors, to either upgrade or replace existing equipment with state-of-the-art equipment, (2) relocate the setpoint and allowable values for the control room airborne radiation monitors to be consistent with the containment airborne radiation monitors TS, and (3) make minor editorial changes to the TS pertaining to the control room airborne radiation monitors and the containment airborne radiation monitors. The proposed changes affect TS Tables 3.3-3, 3.3-4, 3.3-5, 3.3-6, 4.3-2, and 4.3-3.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Control Room Airborne Radiation Monitors

The proposed change would permit relocation of the setpoint and allowable values for the monitors from the Technical Specifications (TSs) to the administrative control procedures. This change is consistent with the existing Containment Airborne Radiation Monitor TSs. This change will not prevent the radiation monitors from performing their intended function following a design basis accident. Therefore, operation of the facility in accordance with this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Containment Area Radiation Monitors

The proposed change deletes the existing Containment Area Radiation Monitors RE-7856-1 and RE-7857-2 and their Engineered Safety Feature Actuation System (ESFAS) function to initiate containment purge isolation on high radiation in containment. The deletion of this ESFAS function does not create a precursor to any analyzed accident since these monitors are for accident mitigation only.

Currently, no release of radioactivity is assumed during a Fuel Handling Accident in containment since the Containment Area Radiation Monitors detect and isolate containment purge prior to release. The proposed deletion will cause some release prior to detection and isolation of purge by the remaining noble gas containment monitors. The consequences of a Fuel Handling Accident inside containment were previously re-evaluated, assuming no containment purge isolation, to resolve inconsistencies in the original analysis assumptions and methodology. The results of the calculation indicated off-site doses well within the limits of 10 CFR 100 and Control Room doses that met the limits of 10 CFR 50 Appendix A General Design Criterion 19. Containment purge isolation on high gaseous activity during a Fuel Handling Accident will still be available with this proposed change but is not required for the dose consequences to remain within the dose criteria. Therefore, the proposed change will not significantly increase the consequences of a Fuel Handling Accident inside containment.

The Loss of Coolant Accident (LOCA) function of the Containment Purge Isolation System (CPIS) signal will be essentially unaffected by this proposed change. Currently, containment purge isolation (containment minipurge) on high radiation signals is a diverse signal with Safety Injection Actuation System (SIAS) and Containment Isolation Actuation System (CIAS). In a LOCA event, containment purge isolation is expected to occur on either SIAS or CIAS prior to a CPIS signal on high radiation in containment. While this proposed change reduces the diversity of radiation monitoring inputs, the diversity of parameters measured (pressure and radiation) is still preserved. Therefore, the proposed change will not increase the consequences of a LOCA.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Control Room Airborne Radiation Monitors

Relocating the monitor setpoint and allowable values from the TSs to the administrative procedures would not alter the design and operational interface between the Control Room Isolation System instrumentation and existing plant equipment. As such, the monitors would continue to operate and perform their intended safety function to isolate the control room following a design basis accident as before. Therefore, operation of the facility in accordance with this proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Containment Area Radiation Monitors

The deletion of the Containment Area Radiation Monitors will not alter the operation of CPIS. The remaining interface between CPIS and existing plant equipment will continue to perform their intended safety function to isolate containment purge by closing the containment purge valves. This function will continue to be performed by Containment Airborne Radiation Monitors 2(3) RT-7804-1 and 2(3) RT-7807-2. Therefore, operation of the facility in accordance with this proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

Control Room Airborne Radiation Monitors

Relocating the monitor setpoint and allowable values to the administrative procedures would not alter the existing margin of safety. The relocation would only relinquish control of the setpoint and allowable values from the TSs to quality-affecting (changes will require a 10 CFR 50.59 evaluation) procedures. Therefore, operation of the facility will not involve a significant reduction in a margin of safety.

Containment Area Radiation Monitors

The proposed change does not affect the margin of safety in Modes 1 through 4 since the diversity of the parameters measured is maintained for minipurge isolation. Either SIAS, CIAS, CPIS, or manual operation will close the containment mini purge valves. The main purge is sealed closed during Modes 1 through 4 with the purge valves closed and deactivated.

The diversity of the parameters measured is not maintained for the containment main purge isolation. The main purge is only applicable during Modes 5 and 6 and main purge isolation is initiated only by either CPIS or manual operation. This proposed change along with the previously submitted PCN-299 reduces the diversity of radiation sensing in containment for CPIS generation from four types (gaseous, iodine, particulate, and gamma) to one type (gaseous activity). Since the consequences of a Fuel Handling Accident inside containment without purge isolation have been calculated to be well within 10 CFR 100 dose limits, the loss of diversity for this accident does not result in a significant reduction in a margin of safety. Therefore, this proposed change will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment requests involve no significant hazards consideration.

Local Public Document Room location: Main Library, University of California, P. O. Box 19557, Irvine, California 92713

Attorney for licensee: T. E. Oubre, Esquire, Southern California Edison Company, P. O. Box 800, Rosemead, California 91770

NRC Project Director: William H. Bateman

Tennessee Valley Authority, Docket No. 50-328, Sequoyah Nuclear Plant, Unit 2, Hamilton County, Tennessee

Date of amendment request: May 19, 1995; revised September 11, 1995 (TS 95-13)

Description of amendment request: The proposed change would revise License Condition 2.C.(17) to extend the required surveillance interval to May 18, 1996, for Surveillance Requirement 4.3.2.1.3. The proposed change would extend the Engineered Safety Features Response Time instrument tests required at 36-month intervals shown in Table 3.3-3 associated with safety injection, feedwater isolation, containment isolation Phase A, auxiliary feedwater pump, essential raw cooling water system, emergency gas treatment system, containment spray, containment isolation Phase B, turbine trip, 6.9-kilovolt shutdown board-degraded voltage or loss of voltage, and automatic switchover to containment sump actuations. The proposed extension will limit the interval past the allowable extension provided by TS 4.0.2 to 5 months.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change is temporary and allows a one-time extension of Surveillance Requirement 4.3.2.1.3 for Cycle 7 to allow surveillance testing to coincide with the seventh refueling outage. The proposed surveillance interval extension will not cause a significant reduction in system reliability nor affect the ability of the systems to perform their design function. Current monitoring of plant conditions and continuation of the surveillance testing required during normal plant operation will continue to be performed to ensure conformance with TS operability requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

Extending the surveillance interval for the performance of specific testing will not create the possibility of a new or different kind of accidents. No changes are required to any system configurations, plant equipment, or analyses. Therefore, this change will not create the possibility of a new or different kind of accident from any previously evaluated.

3. Involve a significant reduction in a margin of safety.

Surveillance interval extensions will not impact any plant safety analyses since the assumptions used will remain unchanged. The safety limits assumed in the accident analyses and the design function of the equipment required to mitigate the consequences of any postulated accidents will not be changed since only the surveillance test interval is being extended. Historical performance generally indicates a high degree of reliability, and surveillance testing performed during normal plant operation will continue to be performed to verify proper performance. Therefore, the plant will be maintained within the analyzed limits, and the proposed extension will not significantly reduce the margin of saety.

The NRC has reviewed the licensee's analysis and, based on thisreview, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11H, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

Virginia Electric and Power Company, Docket Nos. 50-338 and 50-339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: September 1, 1995

Description of amendment request: The proposed changes to the Technical Specifications (TS) for the North Anna Power Station, Units 1&2 (NA-1&2) would allow a single outage of up to 14 days for each emergency diesel generator (EDG) once every 18 months. The purpose of the outage is the performance of a preventive maintenance inspection, appropriate for diesels used for this class of standby service, which requires disassembly of the EDG. Currently this maintenance inspection is performed during refueling outages. The proposed changes would permit this maintenance inspection to be performed during Modes 1 to 4 in addition to the current allowance during Modes 5 or 6.

A probabilistic safety analysis (PSA) has been performed which demonstrates that a fourteen (14) day maintenance inspection outage, once every eighteen (18) months for each EDG, results in no significant change in core damage frequency assuming adequate compensatory measures are in place. The compensatory measures include requirements that the other EDGs, off-site power supply, and the alternate A.C. diesel (AAC DG) be operable during the preventive maintenance inspection outage.

The effect of the proposed change has been calculated to be an increase in core damage frequency of approximately 1E-6 per year, which is not considered to be a significant change (i.e., an acceptable change in risk, or a non-risk significant change) from the baseline core damage frequency of 4.1E-5.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Specifically, operation of North Anna Power Station in accordance with the [proposed] Technical Specifications changes will not:

a. involve a significant increase in the probability or consequences of an accident previously evaluated. The probabilistic safety analysis (PSA) demonstrates that the increase in core damage frequency due to performing the EDG maintenance inspection over a fourteen day period once every 18 months is not significant as long as the AAC DG is operable to act as a source of emergency power to replace the EDG. The period of time during which the EDG is unavailable is short enough to limit the impact of using the manually operated AAC DG as a replacement for the automatically operated EDG.

b. create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed Technical Specifications changes only modify the operability of an EDG for a limited and defined period of time. The UFSAR [Updated Final Safety Analysis Report] accidents are analyzed assuming that the EDG is the worst single failure. This assumption is more severe than the proposed Technical Specifications changes which replaces the EDG with the AAC DG. Similarly, the PSA performed to evaluate the proposed Technical Specifications changes considered all of the initiating events defined for the PSA performed for the Individual Plant Examination. No new initiators were defined as a result of a review of the PSA model. Therefore, it is concluded that no new or different kind of accident from any previously evaluated has been created.

c. The proposed Technical Specifications changes do not result in a reduction in margin of safety as defined in the basis for any Technical Specifications. The PSA was performed to evaluate the concept of a one time outage. The results of the analyses show no significant change in the core damage frequency. As described above the proposed Technical Specifications changes only modify the operability of an EDG for a limited and defined period of time. Thus, operation with slightly increased EDG unavailability due to maintenance, and the AAC DG operable is acceptable.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903-2498.

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: David B. Matthews

Wolf Creek Nuclear Operating Corporation, Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: March 24, 1995, as supplemented by letter dated August 16, 1995.

Description of amendment request: This request proposes to revise Technical Specification 1.7, ''Containment Integrity,'' Technical Specification 3/4.6.1, ''Containment Integrity,'' Technical Specification 3/4.6.3, ''Containment Isolation Valves,'' and their associated Bases. These proposed changes will remove Technical Specification Table 3.6-1 ''Containment Isolation Valves,'' to Wolf Creek Generating Station (WCGS) procedures. This proposed change is in accordance with the guidance provided in Generic Letter 91-08, ''Removal of Component Lists from Technical Specifications,'' dated May 6, 1991. In addition, this request proposes to add a footnote to Technical Specification 3.6.3 extending the allowed outage time for the component cooling water (CCW) system reactor coolant pump seal water supply and return valves. This determination supersedes the staff's proposed no significant hazards consideration determination evaluation for the requested changes that was published on April 26, 1995 (60 FR 20532).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes simplify the technical specifications, meet the regulatory requirements for control of containment isolation, and are consistent with the guidelines of GL 91-08. The procedural details of Technical Specification Table 3.6-1 have not been changed, but only relocated to a different controlling document. The proposed changes are administrative in nature, should result in improved administrative practices, and do not affect plant operations. The addition of the footnote to allow up to 12 hours for valve testing the CCW MOVs [motor-operated valves] does not affect the severity of any accident previously evaluated. This footnote does not impact plant safety since the second isolation device in the affected penetrations would still be available to provide isolation between the RCS and the outside atmosphere.

The probability of occurrence of a previously evaluated accident is not increased because this change does not introduce any new potential accident initiating conditions. The consequences of an accident previously evaluated is not increased because the ability of containment to restrict the release of any fission product radioactivity to the environment will not be degraded by this change.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes are administrative in nature, do not result in physical alterations or changes to the operation of the plant, and cause no change in the method by which any safety-related system performs its function. The addition of the footnote to allow up to 12 hours for valve testing the CCW MOVs does not affect the severity of any accident previously evaluated. The additional time provides assurance that the inoperable valve is in proper working order prior to returning it to OPERABLE status. Therefore, this proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The administrative change to relocate Technical Specification Table 3.6-1 to appropriate plant procedures does not alter the basic regulatory requirements for containment isolation and will not adversely affect containment isolation capability for credible accident scenarios. Adequate control of the content of the table is assured by existing plant procedures. The additional footnote to extend the allowed outage time to 12 hours for the CCW MOVs does not affect containment isolation capability since the second isolation device in the affected penetrations would still be available to provide isolation between the RCS and the outside atmosphere, and to ensure that a release of radioactive material to the environment following an accident will not exceed the assumptions used in the LOCA Analyses.

The proposed relocation of the Technical Specification Table 3.6-1 does not alter the requirements for containment isolation valve operability currently in the technical specifications. The LCO and Surveillance Requirements would be retained in the revised technical specifications. Therefore, the proposed change will not affect the meaning, application, and function of the current technical specification requirements for the valves in Table 3.6-1.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: William H. Bateman

Wolf Creek Nuclear Operating Corporation, Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: August 22, 1995

Description of amendment request: The proposed license amendment request would relocate Technical Specification Tables 3.3-2, ''Reactor Trip System Instrumentation Response Times,'' and 3.3-5, ''Engineered Safety Features Response Times,'' and applicable Bases discussions, to Updated Safety Analysis Report (USAR) Chapter 16. The NRC has already implemented this line-item technical specification improvement in the new Standard Technical Specifications (NUREG-1431 for Westinghouse plants). This amendment request follows the guidance provided by the NRC in Generic Letter 93-08, ''Relocation of Technical Specification Tables of Instrument Response Time Limits,'' for relocating instrument response time tables.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

This license amendment request does not change any Reactor Trip System (RTS) or Engineered Safety Features Actuation System (ESFAS) instrument response times or surveillance intervals currently prescribed in Technical Specification Tables 3.3-2 and 3.3-5. The RTS and ESFAS will continue to function in a manner consistent with the assumptions in the Updated Safety Analysis Report Chapter 15 accident analyses and the plant design basis. Therefore, overall protection system performance will remain within the bounds of the accident analyses documented in USAR Chapter 15. As such, there will be no degradation in system performance, nor will there be an increase in the number of challenges to equipment assumed to function during an accident situation.

The proposed technical specification revision does not involve any hardware changes or changes to any instrumentation setpoints, system operating parameters, or system accident mitigation capabilities, nor do the changes affect the probability of any event initiators. Thus, the proposed change will not result in an increase in the consequences of or the probability of occurrence of any accident or safety-related equipment malfunction.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

As discussed above, there are no hardware changes associated with this proposed amendment request, nor are there any changes in the method by which any safety-related plant system performs its safety function. The normal manner of plant operation is not affected by this proposed change.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes. Therefore, the possibility of a new or different kind of accident is not created by the proposed changes.

3. The proposed change does not involve a significant reduction in a margin of safety.

No response times will be changed by this amendment request. The proposed request only changes the document where the response times will be listed. This proposed amendment request will not affect the manner in which safety limits or limiting safety system settings are determined, nor will there by [be] any effect on plant systems necessary to assure the accomplishment of protection functions. The proposed change will not impact any margin of safety defined in the basis for any Technical Specification.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: William H. Bateman

Previously Published Notices Of Nonsideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and opportunity for a hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the Federal Register on the day and page cited. This notice does not extend the notice period of the original notice.

Indiana Michigan Power Company, Docket No. 50-315, Donald C. Cook, Nuclear Plant, Unit No. 1, Berrien County, Michigan

Date of application for amendment: August 4, 1995 (AEP:NRC:1129E)

Description of amendment request: The proposed amendment would modify Technical Specification 4.4.5.4 and 4.4.5.5, on steam generators, to allow for repair of hybrid expansion joint sleeves under redefined repair boundary limits.

Date of publication of individual notice in the Federal Register: August 14, 1995 (60 FR 41904)

Expiration date of individual notice: For comments: August 29, 1995; hearing requests: September 13, 1995

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: August 21, 1995Description of amendments request: Amend technical specification 3.7.5.c to allow an increase in the average essential raw cooling water supply header temperature from 84.5°F to 87°F until September 30, 1995.

Date of publication of individual notice in the Federal Register: August 28, 1995 (60 FR 44517)

Expiration date of individual notice: September 12, 1995

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

Notice Of Issuance Of Amendments To Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the Federal Register as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

Arizona Public Service Company, et al., Docket Nos. STN 50-528, STN 50-529, and STN 50-530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of application for amendments: June 17, 1994

Brief description of amendments: These amendments revise the surveillance requirement and Bases section of TS 4.7.1.6 to increase the minimum nitrogen accumulator pressure for the atmospheric dump valves (ADVs).

Date of issuance: September 6, 1995

Effective date: September 6, 1995

Amendment Nos.: Unit 1 - Amendment No. 99; Unit 2 - Amendment No. 87; Unit 3 - Amendment No. 70

Facility Operating License Nos. NPF-41, NPF-51, and NPF-74: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 17, 1994 (59 FR 42333) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 6, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Phoenix Public Library, 12 East McDowell Road, Phoenix, Arizona 85004

Arizona Public Service Company, et al., Docket Nos. STN 50-528, STN 50-529, and STN 50-530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of application for amendments: March 31, 1995

Brief description of amendments: The amendments clarify the shutdown margin definition, change the shutdown margin applicability and surveillance requirements to comply with the safety analysis assumptions for subcritical inadvertent control element assembly withdrawal (UFSAR Section 15.4, and expand the applicability for core protection calculator (CPC) operability. In addition, the amendments add a reference to the Core Operating Limits Report for the MODE 6 refueling boron concentration limit. The amendments also change the power calibration requirements for the linear power level, the CPC delta T power, and CPC nuclear power signals to allow more conservative settings than previously required.

Date of issuance: September 1, 1995

Effective date: September 1, 1995

Amendment Nos.: Unit 1 - Amendment No. 98; Unit 2 - Amendment No. 86; Unit - Amendment No. 69

Facility Operating License Nos. NPF-41, NPF-51, and NPF-74: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: June 6, 1995 (60 FR 29871) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 1, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Phoenix Public Library, 1221 N. Central, Phoenix, Arizona 85004

Carolina Power & Light Company, Docket No. 50-261, H. B. Robinson SteamElectric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: June 3, 1995, as supplemented on August 7, 1995. The supplemental submittal did not expand the scope of the original Federal Register notice or change the no significant hazards determination.

Brief description of amendment: The amendment clarifies the definition of operability of the charging pumps by adding a footnote to TS Section 3.2.2.a that states that the connectibility of the emergency power sources is not required for charging pump operability. The bases statement for TS 3.2.2 is also changed for clarification.

Date of issuance: September 5, 1995

Effective date: September 5, 1995

Amendment No.: 166

Facility Operating License No. DPR-23. Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35063) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 5, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550

Commonwealth Edison Company, Docket Nos. STN 50-454 and STN 50-455, Byron Station, Unit Nos. 1 and 2, Ogle County, IllinoisDocket Nos. STN 50-456 and STN 50-457, Braidwood Station, Unit Nos. 1 and 2, Will County, Illinois

Date of application for amendments: February 21, 1995

Brief description of amendments: The amendments revise the technical specifications to permit replacement of the reactor coolant resistance temperature detector (RTD) bypass manifold system with fast response RTDs mounted in thermowells welded directly into the reactor coolant system piping.

Date of issuance: September 5, 1995

Effective date: September 5, 1995

Amendment Nos.: 74 and 66

Facility Operating License Nos. NPF-37, NPF-66, NPF-72 and NPF-77: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35063) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 5, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: For Byron, the Byron Public Library District, 109 N. Franklin, P.O. Box 434, Byron, Illinois 61010; for Braidwood, the Wilmington Public Library, 201 S. Kankakee Street, Wilmington, Illinois 60481

Commonwealth Edison Company, Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendments: February 23, 1995

Brief description of amendments: The amendment revises the Quad Cities Nuclear Power Station, Units 1 and 2, operating licenses to reflect the transfer of the Iowa-Illinois Gas and Electric Company's 25 percent undivided ownership to MidAmerican Energy Company.

Date of issuance: September 11, 1995

Effective date: As of the consummation of the merger between Iowa-Illinois Gas and Electric Company, Midwest Power Systems, Inc., MidAmerican Energy Company, and Midwest Resources, Inc.

Amendment Nos.: 159 and 155

Facility Operating License Nos. DPR-29 and DPR-30: The amendments revised the operating licenses.

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35054) The Commission's related evaluation of the amendments is contained in an Environmental Assessment and Finding of No Significant Impact dated March 21, 1995, and in a Safety Evaluation dated September 11, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021

Commonwealth Edison Company, Docket Nos. 50-295 and 50-304, Zion Nuclear Power Station Units 1 and 2, Lake County, Illinois

Date of application for amendments: May 31, 1995

Brief description of amendments: The amendments authorize an alternative repair criteria for defects found in the tube expansion region within the tubesheet.

Date of issuance: September 11, 1995

Effective date: September 11, 1995

Amendment Nos.: 168 and 155

Facility Operating License Nos. DPR-39 and DPR-48: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35067) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 11, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Waukegan Public Library, 128 N. County Street, Waukegan, Illinois 60085

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of application for amendments: September 19, 1994, as supplemented April 26 and June 19, 1995

Brief description of amendments: These changes to the Technical Specifications (TS) increase the enrichment limits for fuel stored in the fuel pools and establish restricted loading patterns and associated burnup criteria for qualifying fuel in the spent fuel pools. In addition, several administrative changes have been included in order to provide clarity to the TS and bring them more in line with the Standard Technical Specifications format. These changes are as follows: (1) The TS index is changed to add TS 3/4.9.12 and 3/4.9.13, Tables 3.9-1 and 3.9-2 and Figure 3.9-1; (2) TS 3/4.9.12, Spent Fuel Pool (SFP) Boron Concentration is added to establish a boron concentration limit and to establish a Limiting Condition for Operation (LCO) for all modes of operation and to allow the numerical value of the limit to be specified in the Core Operating Limits Report (COLR); (3) TS 3/4.9.13, Tables 3.9-1 and 3.9-2 and Figure 3.9-1 are being added to establish restricted loading patterns for spent fuel storage and associated burnup criteria; (4) Corresponding BASES for TS 3/4.9.12 and 3/4.9.13 are added to explain the basis for each LCO, Action Statement and Surveillance

Requirement covered by the subject TS; (5) TS 5.6, Fuel Storage, is changed to reflect limits for criticality analysis for fuel storage; and (6) TS 6.9, Reporting Requirements, is changed to reflect the inclusion of the SFP boron concentration limit values in the COLR as established by TS 3/4.9.12.

Date of issuance: August 31, 1995

Effective date: As of the date of issuance to be implemented within 30 days

Amendment Nos.: 134 and 128

Facility Operating License Nos. NPF-35 and NPF-52: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: May 23, 1995 (60 FR 27338) The June 19, 1995, letter provided clarifying information that did not change the scope of the September 19, 1994, application and the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 31, 1995, and Environmental Assessment dated August 15, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

Duquesne Light Company, et al., Docket Nos. 50-334 and 50-412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of application for amendments: August 31, 1994, as supplemented May 18, 1995.

Brief description of amendments: These amendments delete Beaver Valley Power Station, Unit 2, License Conditions 2.C.(3), 2.C.(5), 2.C.(7), 2.C.(8), 2.C.(9) and 2.C.(10) to reflect completion of activities required by these license conditions and make the following revisions to the Beaver Valley Power Station, Units 1 and 2, TSs:

1. Eliminate references to specific frequencies for each of the TS required audits (TS 6.2.2.8).

2. Eliminate references to reviews and audits of the Emergency plan and Security Plant (TSs 6.5.2.8 and 6.8.1).

3. Include Offsite Dose Calculation Manual and Process Control Program and associated implementing procedures into the list of required audits (TS 6.5.2.8).

4. Editorial changes which were necessitated by a reorganization (TS 6.2.1, 6.2.3.1, 6.2.3.4, 6.5.2.2, 6.5.2.8, 6.5.2.9, and 6.5.2.10).

5. Eliminate reference to Appendix A of 10 CFR Part 55 (TS 6.4.1).

6. Separate the Inservice Inspection (ISI) and Inservice Testing (IST) Programs surveillance requirements and remove the requirement that relief requests be granted before they are implemented for both IST and ISI (TS 4.0.5).

The May 18, 1995, letter requested withdrawal of the proposed changes to TS 6.5.2.8 dealing with audits of the Beaver Valley Power Station, Units 1 and 2, fire protection program and withdrawal of a proposed 25-percent grace period for all audit frequencies (Item 6 in August 31, 1994 application).

Date of issuance: August 31, 1995

Effective date: Units 1 and 2, as of the date of issuance and shall be implemented within 60 days.

Amendment Nos.: 191 and 74

Facility Operating License Nos. DPR-66 and NPF-73: Amendments revised the Units 1 and 2 Technical Specifications, and the Unit 2 License.

Date of initial notice in Federal Register: (59 FR 65812) December 21, 1994. The May 18, 1995, letter did not change the original no significant hazards consideration determination or expand the scope of the December 21, 1994, Federal Register notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 31, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001

Duquesne Light Company, et al., Docket Nos. 50-334 and 50-412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of application for amendments: October 11, 1994, as supplemented June 23, 1995, and August 24, 1995

Brief description of amendments: These amendments revise Beaver Valley Power Station Technical Specifications (TSs) 1.18, ''Quadrant Power Tilt Ratio,'' 3/4.2.4, ''Quadrant Power Tilt Ratio,'' the table Notation of TS Table 3.3-1, ''Reactor Trip System Instrumentation,'' and associated Bases to incorporate the guidance provided in the NRC's Improved Standard Technical Specifications (NUREG-1431, Revision 1) to these TSs. The amendments clarify the requirements of the subject TSs with regard to the use of excore power range neutron flux detectors to monitor quadrant power tilt ratio when an excore power range neutron flux instrument is inoperable. The changes also make several minor editorial changes in the subject TSs.

Date of issuance: September 15, 1995

Effective date: As of date of issuance, to be implemented within 60 days.

Amendment Nos.: 192 and 75

Facility Operating License Nos. DPR-66 and NPF-73: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39436) The August 24, 1995, letter provided typed final TS pages, with minor editorial changes, for issuance of these amendments. The August 24, 1995, letter did not change the initial proposed no significant hazards consideration determination or expand the scope of the August 2, 1995, Federal Register notice. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 15, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001

Entergy Operations, Inc., Docket No. 50-382, Waterford Steam ElectricStation, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: June 22, 1994

Brief description of amendment: The amendment changes the Appendix A Technical Specifications by removing the seismic and meteorological monitoring instrumentation requirements. These requirements are to be relocated in the Updated Final Safety Analysis Report.

Date of issuance: September 5, 1995

Effective date: September 5, 1995

Amendment No.: 112

Facility Operating License No. NPF-38. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 3, 1994 (59 FR 39585) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 5, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122

Entergy Operations, Inc., Docket No. 50-382, Waterford Steam ElectricStation, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: June 22, 1994, and December 9, 1994

Brief description of amendment: The amendment changes the Appendix A TSs by revising the plant protection system trip setpoints and allowable values such that they will be consistent with the current setpoint/uncertainty methodology being implemented at Waterford 3.

Date of issuance: September 5, 1995

Effective date: September 5, 1995

Amendment No.: 113

Facility Operating License No. NPF-38. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 3, 1994 (59 FR 39586) and February 1, 1995 (60 FR 6300)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 5, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122

Entergy Operations, Inc., Docket No. 50-382, Waterford Steam ElectricStation, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: December 9, 1994, as supplemented by letter dated July 25, 1995

Brief description of amendment: The requested changes revised the allowable opening tolerances on the pressurizer safety valves (PSVs) and the main steam line safety valves (MSSVs) from plus or minus 1 percent to plus or minus 3 percent. However, following testing, the as-left lift setting of the PSVs and MSSVs will be within plus or minus 1 percent of the pressure specified in the Technical Specifications.

Date of issuance: September 11, 1995

Effective date: September 11, 1995

Amendment No.: 111

Facility Operating License No. NPF-38. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: February 1, 1995 (60 FR 6300) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 11, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, Louisiana 70122

Florida Power and Light Company, Docket No. 50-335, St. Lucie Plant, Unit No. 1, St. Lucie County, Florida

Date of application for amendment: May 23, 1994

Brief description of amendment: The amendment revises Technical Specification 3.5.2 for Emergency Core Cooling Systems (ECCS) by removing the option that allows High Pressure Safety Injection (HPSI) Pump 1C to be used as an alternative to the preferred pump for subsystem operability. HPSI pump 1C is an installed spare which is not required to be maintained in an operable status, and this change upgrades the ECCS operability requirements consistent with actual plant operating needs.

Date of issuance: September 11, 1995

Effective date: September 11, 1995

Amendment No.: 139

Facility Operating License No. DPR-67: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 6, 1994 (59 FR 34663) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 11, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

Florida Power and Light Company, et al., Docket No. 50-389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of application for amendment: February 27, 1995

Brief description of amendment: This amendment will change Table 3.3-3 and 3.3-4 to accommodate an improved coincidence logic and relay replacement for the 4.16 kV Loss of Voltage Relays. Actions required for certain trip units with the number of operable channels one less than the total number of channels will also be changed. In addition, the format used to state the time delay for the 4.16 kV Degraded Voltage trip unit will be revised.

Date of issuance: September 1, 1995

Effective date: September 1, 1995

Amendment No.: 79

Facility Operating License No. NPF-16: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: March 29, 1995 (60 FR 16187) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 1, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-321 and 50-366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of application for amendments: January 13, 1995, as supplemented by letters dated April 5 and June 20, 1995.

Brief description of amendments: The amendments modify

Facility Operating License Nos. DRP-57 and NPF-5 and the corresponding TS for Hatch Units 1 and 2, respectively, to authorize an increase in the maximum power level from 2436 megawatts thermal (MWt) to 2558 MWt. The amendments also approve changes to the Technical Specification to implement uprated power operation.

Date of issuance: August 31, 1995

Effective date: As of the date of issuance to be implemented prior to startup in Cycle 17 for Unit 1; and prior to startup in Cycle 13 for Unit 2

Amendment Nos.: 197 and 138

Facility Operating License Nos. DPR-57 and NPF-5. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35072) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated August 31, 1995 and an Environmental Assessment dated July 21, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-321 and 50-366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of application for amendments: June 6, 1995, as supplemented August 9, 1995.

Brief description of amendments: The amendments revise Technical Specification Surveillance Requirements (SR) 3.6.4.1.3 and 3.6.4.1.4 for the secondary containment drawdown. The revision reduces the SR acceptance criteria to greater than or equal to 0.20 inch water gauge (wg) negative pressure from greater than or equal to O.25 inch wg negative pressure. The appropriate TS Bases pages are also changed to reflect the TS revision.

Date of issuance: September 11, 1995

Effective date: As of the date of issuance to be implemented within 60 days

Amendment Nos.: 198 and 139

Facility Operating License Nos. DPR-57 and NPF-5. Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: June 21, 1995 (60 FR 32364) The August 9, 1995, letter provided clarifying information that did not change the scope of the June 6, 1995, application and the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 11, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513

GPU Nuclear Corporation, et al., Docket No. 50-219, Oyster Creek Nuclear Generating Station, Ocean County, New Jersey

Date of application for amendment: June 26, 1995

Brief description of amendment: The amendment revises the snubber visual inspection intervals to match the schedule developed by the NRC staff for use with a 24-month refueling interval. This schedule was documented in Generic Letter 90-09. The amendment also revises the bases for the snubber visual inspection interval to be consistent with the bases described in Generic Letter 90-09.

Date of issuance: September 6, 1995

Effective date: September 6, 1995

Amendment No.: 182

Facility Operating License No. DPR-16. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39440). The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated September 6, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Ocean County Library, Reference Department, 101 Washington Street, Toms River, NJ 08753

Gulf States Utilities Company, Cajun Electric Power Cooperative, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: May 25, 1995

Brief description of amendment: The amendment revises the Physical Security Plan vital island requirements.

Date of issuance: September 12, 1995

Effective date: September 12, 1995

Amendment No.: 83

Facility Operating License No. NPF-47. The amendment revised the operating license.

Date of initial notice in Federal Register: July 19, 1995 (60 FR 37091) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 12, 1995.No significant hazards consideration comments received. No

Local Public Document Room location: Government Documents Department, Louisiana State University, Baton Rouge, LA 70803

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: May 25, 1995, as supplemented by letter dated August 3, 1995.

Brief description of amendments: The amendments revised the technical specifications (TSs) on containment leakage, making the action statement consistent with the need to perform Type C testing at power, and replacing the surveillance requirements with a single requirement to apply the requirements of Appendix J as modified by approved exemptions. The amendments also revised the TSs on containment integrity, containment leakage, and containment air locks, to eliminate the numerical value of calculated peak containment internal pressure related to the design basis accident.

Date of issuance: September 7, 1995

Effective date: September 7, 1995

Amendment Nos.: Unit 1 - Amendment No. 80; Unit 2 - Amendment No. 69

Facility Operating License Nos. NPF-76 and NPF-80. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: July 19, 1995 (60 FR 37092) The August 3, 1995, supplement provided clarifying information and did not change the original no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 7, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, Texas 77488

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: May 31, 1995, as supplemented by letter dated August 2, 1995

Brief description of amendments: The amendments modified (by relocation to the Technical Requirements Manual) TS 3/4.1.2.1, Boration Systems/Flow Paths - Shutdown, TS 3/4.1.2.2, Boration Systems/Flow Paths - Operating, TS 3/4.1.2.3, Charging Pumps - Shutdown, TS 3/4.1.2.4, Charging Pumps - Operating, TS 3/4.1.2.5, Borated Water Sources - Shutdown, TS 3/4.1.2.6, Borated Water Sources - Operating, TS 3/4.4.2.1, Safety Valves - Shutdown, and the associated Bases.

Date of issuance: September 5, 1995

Effective date: September 5, 1995

Amendment Nos.: Unit 1 - Amendment No. 79; Unit 2 - Amendment No. 68

Facility Operating License Nos. NPF-76 and NPF-80. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39441) The additional information contained in the supplemental letter dated August 2, 1995, was clarifying in nature and thus, within the scope of the initial notice and did not affect the staff's proposed no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 5, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488

Indiana Michigan Power Company, Docket No. 50-315, Donald C. Cook, Nuclear Plant, Unit No. 1, Berrien County, Michigan

Date of application for amendment: February 3, 1995, as supplemented April 25, 1995.

Brief description of amendment: The amendment modifies the technical specifications to extend the interim steam generator tube plugging criteria used in Cycle 14 to the next operating cycle (Cycle 15).

Date of issuance: September 13, 1995

Effective date: September 13, 1995

Amendment No.: 200

Facility Operating License No. DPR-58. Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: July 19, 1995 (60 FR 37093) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 13, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085

Indiana Michigan Power Company, Docket No. 50-316, Donald C. Cook, Nuclear Plant, Unit No. 2, Berrien County, Michigan

Date of application for amendment: February 15, 1994, as supplemented June 29, 1995

Brief description of amendment: The amendment deletes Technical Specification section 3/4.3.4, associated bases, and associated index listings for the Unit 2 turbine overspeed protection system.

Date of issuance: September 1, 1995

Effective date: September 1, 1995

Amendment No.: 185

Facility Operating License No. DPR-74. Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: March 30, 1994 (59 FR 14890) The licensee's submittal of June 29, 1995, did not change the basis for the proposed no significant hazards consideration determination.The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 1, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085

Nebraska Public Power District, Docket No. 50-298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: June 15, 1995

Brief description of amendment: The amendment changes the Technical Specifications to revise the definition for logic system functional test and revises the surveillance interval for emergency core cooling system logic system functional testing from 6 months to 18 months.

Date of issuance: September 7, 1995

Effective date: September 7, 1995

Amendment No.: 171

Facility Operating License No. DPR-46. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 19, 1995 (60 FR 37096) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 7, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Auburn Public Library, 118 15th Street, Auburn, NE 68305

Niagara Mohawk Power Corporation, Docket No. 50-410, Nine Mile PointNuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: January 6, 1995

Brief description of amendment: The amendment incorporates Limiting Condition for Operation 3.3.3.1 from Standard Technical Specifications into Technical Specification (TS) 3/4.3.7.5, Accident Monitoring Instrumentation and make associated changes in TS 3/4.4.2, Safety Relief Valves.

Date of issuance: September 11, 1995

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 69

Facility Operating License No. NPF-69: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: February 15, 1995 (60 FR 8748)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 11, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126

Niagara Mohawk Power Corporation, Docket No. 50-410, Nine Mile PointNuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: January 6, 1995, as supplemented April 18, 1995

Brief description of amendment: The amendment revises Technical Specifications (TSs) Sections 3.8.1.1 and 3.8.1.2; TS Surveillance Requirements Section 4.8.1.1.2; TS Bases Section 3/4.8.1.3; and TS Administrative Controls Section 6.8.4. The changes include: updating the minimum day tank and storage tank oil inventory, specific actions required if oil level fall below minimum required, revising and relocating the fuel oil sampling and testing criteria to the associated Bases, and specific action to be taken if the fuel oil properties do not meet the specified limits. In addition, a requirement was added for a diesel fuel oil testing program. These changes are consistent with guidance provided in NUREG-1434.

Date of issuance: September 15, 1995

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 70

Facility Operating License No. NPF-69: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: February 15, 1995 (60 FR 8747) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 15, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126

North Atlantic Energy Service Corporation, Docket No. 50-443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: April 16, 1995.

Description of amendment request: The amendment revises the Appendix A Technical Specifications (TS) relating to containment building penetrations. Specifically, the amendment modifies Limiting Conditions for Operation 3.9.4 to permit both doors of one personnel airlock to be open during core alterations or irradiated fuel movement if certain conditions are met and to add equivalent and alternate penetration closure methodologies. Surveillance Requirement 4.9.4 is changed to reflect that the penetrations are to be verified to be in the condition required. Bases Section 3/4 9.4 also is revised to reflect the changes described above.

Date of issuance: August 31, 1995

Effective date: As of the date of issuance, to be implemented within 60 days.

Amendment No.: 40

Facility Operating License No. NPF-86. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: June 21, 1995 (60 FR 32369) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 31, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833

North Atlantic Energy Service Corporation, Docket No. 50-443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: May 30, 1995.

Description of amendment request: The amendment revises the Appendix A Technical Specifications (TS) relating to Moderator Temperature Coefficient. The amendment changes the upper limit for the moderator temperature coefficient (MTC) for certain operating conditions. Additionally, a reference for the analytical method used to determine the cycle-specific MTC upper limit is added to TS 6.8.1.6.b.

Date of issuance: September 14, 1995

Effective date: As of the date of issuance to be implemented within 60 days.

Amendment No.: 41

Facility Operating License No. NPF-86. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35082). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 14, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833

North Atlantic Energy Service Corporation, Docket No. 50-443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: June 16, 1995

Description of amendment request: The amendment revises the Appendix A Technical Specifications (TS) relating to core reactivity control available from borated water sources. The amendment changes the minimum boron concentration specified for the refueling water storage tank (RWST) in Limiting Condition for Operation (LCO) in TS 3.1.2.5 and replaces the minimum specified concentration for boron with an acceptable range of boron concentration for the RWST and the accumulators in the LCOs for TS 3.1.2.6, 3.5.1.1, and 3.5.4.

Date of issuance: September 14, 1995

Effective date: As of the date of issuance, to be implemented prior to entering MODE 4 following the fourth refueling outage.

Amendment No.: 42

Facility Operating License No. NPF-86. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39442). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 14, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833

Northeast Nuclear Energy Company, Docket No. 50-245, MillstoneNuclear Power Station, Unit 1, New London County, Connecticut

Date of application for amendment: July 11, 1995

Brief description of amendment: The amendment modifies Technical Specification 3.5.F.7 to also allow the use of pull-to-lock switches to defeat the automatic initiation of the emergency core cooling system while in the refuel condition. The amendment also makes editorial corrections and makes changes to the associated Bases section.

Date of issuance: September 13, 1995

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 88

Facility Operating License No. DPR-21. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39442). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 13, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360

Northeast Nuclear Energy Company, Docket No. 50-245, MillstoneNuclear Power Station, Unit 1, New London County, Connecticut

Date of application for amendment: July 18, 1995

Brief description of amendment: The amendment adds operability and surveillance requirements for reactor pressure vessel overfill protection instrumentation. The amendment also adds the associated Bases.

Date of issuance: September 13, 1995

Effective date: As of the date of issuance to be implemented within 60 days.

Amendment No.: 87

Facility Operating License No. DPR-21. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39443) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 13, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360

Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station,Unit No. 1, Washington County, Nebraska

Date of amendment request: April 7, 1995

Brief description of amendment: The amendment revises the technical specifications (TS) to relocate the axial power distribution limits to the Core Operating Limits Report (COLR).

Date of issuance: September 1, 1995

Effective date: September 1, 1995

Amendment No.: 170

Facility Operating License No. DPR-40. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: May 23, 1995 (60 FR 27339) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 1, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102

Omaha Public Power District, Docket No. 50-285, Fort Calhoun Station,Unit No. 1, Washington County, Nebraska

Date of amendment request: May 8, 1995, as supplemented by letter dated July 11, 1995.

Brief description of amendment: The amendment changes Sections 2.3, 3.1, 3.2, 3.3, and 3.6 of the Technical Specifications in accordance with the guidance of Generic Letter (GL) 93-05, ''Line Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation,'' dated September 27, 1993. The changes are consistent with Station operating experience and NUREG-1366, ''Improvements to Technical Specifications Surveillance Requirements,'' dated December 1992. In addition, a change was made to TS Section 3.1 in accordance with the Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors. Also, changes were made to the TS sections identified above for clarity and to correct administrative errors.

Date of issuance: September 7, 1995

Effective date: September 7, 1995

Amendment No.: 171

Facility Operating License No. DPR-40. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: June 6, 1995 (60 FR 29883) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 7, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: W. Dale Clark Library, 215 South 15th Street, Omaha, Nebraska 68102

Pennsylvania Power and Light Company, Docket No. 50-387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of application for amendment: April 11, 1995

Brief description of amendment: This amendment extends on a one-time basis the allowed outage time from 3 to 7 days for one offsite circuit being out of service.

Date of issuance: August 31, 1995

Effective date: As of the date of issuance and is to be implemented within 30 days.

Amendment No.: 153

Facility Operating License No. NPF-14: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: June 6, 1995 (60 FR 29886). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 31, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of application for amendments: February 2, 1995

Brief description of amendments: These amendments change the Technical Specifications for the two Susquehanna units to increase the licensed discharge fuel assembly for SPC 9X9-2 fuel from 40 to 45 GWD/MTU. This change is consistent with the Commissions approval of Topical Report PL-NF-94-005-P, ''Technical Basis for SPC 9X9-2 Extended Fuel Exposure at Susquehanna SES,'' documented in a letter to PP&L dated December 15, 1994.

Date of issuance: September 12, 1995

Effective date: September 12, 1995

Amendment Nos.: 154 and 124

Facility Operating License Nos. NPF-14 and NPF-22. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: March 29, 1995 (60 FR 16194) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 12, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701

Pennsylvania Power and Light Company, Docket No. 50-388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of application for amendment: March 31, 1995

Brief description of amendment: This amendment changes Technical Specification Section 6.9.3.2 to allow four GE demonstration assemblies to be loaded into Susquehanna Unit 2, Cycle 8 core.

Date of issuance: September 13, 1995

Effective date: As of date of issuance and shall be implemented within 30 days.

Amendment No.: 125

Facility Operating License No. NPF-22. This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: April 26, 1995 (60 FR 20523) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 13, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701

Pennsylvania Power and Light Company, Docket No. 50-388, Susquehanna Steam Electric Station, Unit 2, Luzerne County, Pennsylvania

Date of application for amendment: May 5, 1995, and supplemented by letter dated August 18, 1995

Brief description of amendment: This amendment deletes from SSES Technical Specification Table 3.6.3-1, ''Primary Containment Isolation Valves,'' three relief valves in the residual heat removal system. These specific valves which were originally intended to support the steam condensing mode, were previously eliminated from the plant design. The valves are being replaced during the September Unit 2 refueling outage and will be replaced by blind flanges.

Date of issuance: September 11, 1995

Effective date: September 11, 1995

Amendment No.: 123

Facility Operating License No. NPF-22. This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: July 5, 1995 (60 FR 35083 and July 17, 1995 (60 FR 36449)The supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original Federal Register notice. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 11, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701

Power Authority of The State of New York, Docket No. 50-286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: February 23, 1995, as supplemented July 28, 1995

Brief description of amendment: The amendment revised the minimum emergency diesel generator (EDG) fuel oil requirements, as indicated in Technical Specification (TS) Section 3.7 (Auxiliary Electrical Systems), from 7056 to 6671 gallons. The actual minimum fuel oil level had always been 6671 gallons; however, the previous TS limit of 7056 gallons was based on a level indicator that had an accuracy of +/- 385 gallons. This revision clarified the TS such that any level indicator can now be used as long as an actual minimum level of 6671 gallons is assured.

Date of issuance: August 30, 1995

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 161

Facility Operating License No. DPR-64: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: March 29, 1995 (60 FR 16196) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 30, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610

Power Authority of The State of New York, Docket No. 50-286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: March 2, 1995

Brief description of amendment: The amendment revised the titles of several management positions as described in Technical Specifications Section 6.0 (Administrative Controls). Specifically, the title of Executive Vice President and Chief Nuclear Officer and the title of Shift Supervisor were changed to Chief Nuclear Officer and Shift Manager, respectively. In addition, the position titles of Senior Reactor Operator and Reactor Operator were deleted and replaced with qualification requirements.

Date of issuance: August 31, 1995

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 162

Facility Operating License No. DPR-64: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: March 29, 1995 (60 FR 16197) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 31, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of application for amendment: October 3, 1994

Brief description of amendment: The amendment proposed changes to FitzPatrick TSs which will extend the instrumentation functional test interval and allowable out-of-service times, remove the average power range monitor downscale scram function and the instrument response time values, and incorporate several editorial, clarification, and correction changes.

Date of issuance: September 11, 1995

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 227

Facility Operating License No. DPR-59: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: November 9, 1994 (59 FR 55887) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 11, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126

Public Service Electric & Gas Company, Docket No. 50-354, Hope Creek Generating Station, Salem County, New Jersey

Date of application for amendment: September 29, 1994

Brief description of amendment: This amendment revises Table 4.3.6-1, ''Control Rod Block Instrumentation Surveillance,'' of the Hope Creek TS. The channel calibration frequencies for the Source Range Monitor (SRM) and the Intermediate Range Monitor (IRM), in TS Table 4.3.6-1, are changed for the up-scale and the down-scale trip functions on each instrument from ''SA'' (once-per-184 days) to ''R'' (once-per-refueling cycle).

Date of issuance: September 12, 1995

Effective date: As of the date of issuance and shall be implemented within 60 days.

Amendment No.: 78

Facility Operating License No. NPF-57: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: June 18, 1995 (60 FR 3676). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 12, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Pennsville Public Library, 190 S. Broadway, Pennsville, New Jersey 08070

Public Service Electric & Gas Company, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: May 2, 1995

Brief description of amendments: The amendments eliminate the monthly manual initiation of auxiliary feedwater from Technical Specification Tables 3.3-3, 3.3.-4 and 4.3-2.

Date of issuance: September 6, 1995

Effective date: Units 1 and 2, as of the date of issuance, to be implemented within 60 days.

Amendment Nos. 175 and 156

Facility Operating License Nos. DPR-70 and DPR-75. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: June 6, 1995 (60 FR 29887)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 6, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079

South Carolina Electric & Gas Company, South Carolina Public ServiceAuthority, Docket No. 50-395, Virgil C. Summer Nuclear Station, Unit No. 1, Fairfield County, South Carolina

Date of application for amendment: June 19, 1995

Brief description of amendment: The amendment restructures the primary containment and primary containment leakage technical specifications to reduce the repetition of those requirements contained in NRC regulations such as Appendix J to 10 CFR Part 50.

Date of issuance: September 5, 1995

Effective date: September 5, 1995

Amendment No.: 126

Facility Operating License No. NPF-12: Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: July 19, 1995 (60 FR 37099)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 5, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Fairfield County Library, 300 Washington Street, Winnsboro, SC 29180

Southern California Edison Company, et al., Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station, Unit Nos. 2 and 3, San Diego County, California

Date of application for amendments: May 20, 1994

Brief description of amendments: The amendments revise Technical Specification 3/4.7.3, ''Component Cooling Water System,'' and the corresponding Bases to support the addition of the component cooling water surge tank backup nitrogen supply (BNS) system.

Date of issuance: September 13, 1995

Effective date: September 13, 1995

Amendment Nos.: Unit 2 - Amendment No. 125; Unit 3 - Amendment No. 114

Facility Operating License Nos. NPF-10 and NPF-15: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 31, 1994 (59 FR

45034)The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated September 13, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Main Library, University of California, P.O. Box 19557, Irvine, California 92713

Tennessee Valley Authority, Docket Nos. 50-259, 50-260, and 50-296, Browns Ferry Nuclear Plant, Units 1, 2, and 3, Limestone County, Alabama

Date of application for amendments: March 31, 1995, supplemented July 14, 1995 (TS 349)

Brief description of amendment: These amendments revise the Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 reactor vessel pressure-temperature curves and bolt-up temperatures.

Date of issuance: September 13, 1995

Effective date: September 13, 1995

Amendment Nos.: 224, 239, 198

Facility Operating License Nos. DPR-33, DPR-52 and DPR-68: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: June 6, 1995 (60 FR 29888)The July 14, 1995 letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 13, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Athens Public library, South Street, Athens, Alabama 35611

Tennessee Valley Authority, Docket Nos. 50-259, 50-260, and 50-296, Browns Ferry Nuclear Plant, Units 1, 2, and 3, Limestone County, Alabama

Date of application for amendments: May 11, 1995, supplemented June 30, 1995 (TS 359)

Brief description of amendment: The amendments provide for the addition of a reactor trip on low scram pilot air header pressure for BFN Unit 3, and revise a note regarding instrumentation requirements for all three BFN reactors.

Date of issuance: August 29, 1995

Effective date: August 29, 1995

Amendment Nos.: 223, 228 and 197

Facility Operating License Nos. DPR-33, DPR-52 and DPR-68: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: June 6, 1995 (60 FR 29889)The June 30, 1995 letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 29, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Athens Public library, South Street, Athens, Alabama 35611

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: June 29, 1995 (TS 95-14)

Brief description of amendments: The amendments revise Technical Specification 3.9.4, Containment Building Penetrations, to allow both sets of containment personnel airlock doors to be open during core alterations and fuel movement provided one door is capable of closure and one train of auxiliary building gas treatment remains operable.

Date of issuance: September 6, 1995

Effective date: September 6, 1995

Amendment Nos.: 209 and 199

Facility Operating License Nos. DPR-77 and DPR-79: Amendments revise the technical specifications.

Date of initial notice in Federal Register: July 19, 1995 (60 FR 37100)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 6, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Chattanooga-Hamilton County Library,1101 Broad Street, Chattanooga, Tennessee 37402

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: August 7, 1995 (TS 95-11)

Brief description of amendments: The amendments revise the time constant used in the overtemperature delta temperature and the overpower delta temperature trip equations of Technical Specification Table 2.2-1.

Date of issuance: September 15, 1995

Effective date: September 15, 1995

Amendment Nos.: 211 and 201

Facility Operating License Nos. DPR-77 and DPR-79: Amendments revise the technical specifications.

Date of initial notice in Federal Register: April 26, 1995 (60 FR 20527); superseded August 15, 1995 (60 FR 42187) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 15, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Chattanooga-Hamilton County Library,1101 Broad Street, Chattanooga, Tennessee 37402

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of application for amendments: August 21, 1995 (TS 95-21)

Brief description of amendments: The amendments change Technical Specification 3.7.5.c to allow an increase in the average essential raw cooling water supply header temperature from 84.5°F to 87°F untilSeptember 30, 1995.

Date of issuance: September 13, 1995

Effective date: September 13, 1995

Amendment Nos.: 210 and 200

Facility Operating License Nos. DPR-77 and DPR-79: Amendments revise the technical specifications.Public comments requested as to proposed no significant hazards consideration: Yes (August 28, 1995, 60 FR 44517). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards determination. No comments have been received. The notice also provided an opportunity to request a hearing, by September 12, 1995, but indicated that if the Commission makes a final no significant hazards consideration determination before the expiration of the notice period, any such hearing would take place after issuance of the amendments.The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated September 13, 1995.

Local Public Document Room location: Chattanooga-Hamilton County Library,1101 Broad Street, Chattanooga, Tennessee 37402

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50-440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of application for amendment: April 28, 1995

Brief description of amendment: The amendment extends for one additional operating cycle the exception to Limiting Condition for Operation 3.0.4 as it applies to the main steam isolation valve leakage control system Technical Specification.

Date of issuance: September 8, 1995

Effective date: September 8, 1995

Amendment No.: 71

Facility Operating License No. NPF-58: This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: May 23, 1995 (60 FR 27344)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 8, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, Ohio 44081

Toledo Edison Company, Centerior Service Company, and The Cleveland Electric Illuminating Company, Docket No. 50-346, Davis-Besse Nuclear Power Station, Unit No. 1, Ottawa County, Ohio

Date of application for amendment: April 10, 1995

Brief description of amendment: This amendment changes auxiliary feedwater system, motor driven feedwater pump, and condensate system Technical Specifications to increase clarity and changes format to more closely follow improved standard technical specifications and increases content of Bases discussions.

Date of issuance: September 5, 1995

Effective date: September 5, 1995

Amendment No.: 200

Facility Operating License No. NPF-3. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39453) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 5, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: University of Toledo Library, Documents Department, 2801 Bancroft Avenue, Toledo, Ohio 43606

Virginia Electric and Power Company, Docket Nos. 50-280 and 50-281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia.

Date of application for amendments: November 10, 1994

Brief description of amendments: Clarify the surveillance requirementsfor the Reactor Protection and Engineered Safeguards Systems instrumentation and actuation logic.

Date of issuance: September 14, 1995

Effective date: September 14, 1995

Amendment Nos.: 205 and 205

Facility Operating License Nos. DPR-32 and DPR-37: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: April 12, 1995 (60 FR 18630) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 14, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Swem Library, College of William and Mary, Williamsburg, Virginia 23185

Virginia Electric and Power Company, Docket Nos. 50-280 and 50-281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia.

Date of application for amendments: July 14, 1995

Brief description of amendments: These amendments would provide a 2-hour allowed outage time for one residual heat removal loop to accommodate plant safety and emergency power systems surveillance testing, permit depressurizing safety injection accumulators in lieu of accumulator isolation, and make administrative changes.

Date of issuance: September 1, 1995

Effective date: September 1, 1995

Amendment Nos.: 204 and 204

Facility Operating License Nos. DPR-32 and DPR-37: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: August 2, 1995 (60 FR 39455) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated September 1, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Swem Library, College of William and Mary, Williamsburg, Virginia 23185

Wisconsin Public Service Corporation, Docket No. 50-305, Kewaunee NuclearPower Plant, Kewaunee County, Wisconsin

Date of application for amendment: December 16, 1994.

Brief description of amendment: The amendment revises the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications Sections 3.4 and 4.1 by removing the limiting conditions for operation (LCO) and the surveillance requirements for the turbine overspeed protection system (TOPS). The TOPS requirements will be relocated to the Updated Safety Analysis Report (USAR).

Date of issuance: August 31, 1995

Effective date: August 31, 1995

Amendment No.: 121

Facility Operating License No. DPR-43. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: January 18, 1995 (60 FR 3676). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 31, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: University of Wisconsin Library Learning Center, 2420 Nicolet Drive, Green Bay, Wisconsin 54301

Notice Of Issuance Of Amendments To Facility Operating Licenses And Final Determination Of No Significant Hazards Consideration And Opportunity For A Hearing (Exigent Public Announcement Or Emergency Circumstances)

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a Federal Register notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By October 27, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear

Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

Niagara Mohawk Power Corporation, Docket No. 50-410, Nine Mile PointNuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: August 28, 1995

Brief description of amendment: The amendment revises Primary Containment Purge System Technical Specification Section 3.6.1.7, Limiting Condition for Operation. The revision extends the amount of time the 12-inch and 14-inch purge system supply and exhaust lines may be used for venting or purging from 90 to 135 hours per 365 days. In addition, expired footnotes were deleted as an editorial change and the associated Bases section was revised.

Date of issuance: August 31, 1995

Effective date: As of the date of issuance to be implemented upon receipt.

Amendment No.: 68

Facility Operating License No. NPF-69: Amendment revises the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: NoThe Commission's related evaluation of the amendment, emergency circumstances and consultation with the State, and final determination of no significant hazards consideration are contained in a Safety Evaluation dated August 31, 1995.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston & Strawn, 1400 L Street, NW., Washington, DC 20005-3502.

NRC Project Director: Ledyard B. Marsh

For the Nuclear Regulatory Commission

John N. Hannon,

Acting Director, Division of Reactor Projects - III/IV Office of Nuclear Reactor Regulation

[Doc. 95–23806 Filed 9–26–95; 8:45 am]

BILLING CODE 7590–01–F

[Docket Nos. STN 50–454, STN 50–455, STN 50–456 and STN 50–457]

Commonwealth Edison Company; Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed no Significant Hazards Consideration Determination, and Opportunity for A Hearing

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of amendments to Facility Operating License Nos. NPF–37, NPF–66, NPF–72, and NPF–77, issued to Commonwealth Edison Company for operation of the Byron Station, Units 1 and 2, located in Ogle County, Illinois and Braidwood Station, Units 1 and 2, located in Will County, Illinois.

The proposed amendments would revise the present voltage-based repair criteria in the Byron 1 and Braidwood 1 Technical Specifications (TSs). These proposed revisions would raise the lower voltage limit from its present value of 1.0 volt to 3.0 volts; there would no longer be an upper voltage limit.

The Braidwood 1 TSs were revised by License Amendment No. 54, issued on August 18, 1994, to add voltage-based repair criteria to the existing steam generator (SG) tube repair criteria. The Byron 1 TSs were revised in a similar manner by License Amendment No. 66, issued on October 24, 1994.

The voltage-based repair criteria in the subject TSs are applicable only to a specific type of SG tube degradation which is predominantly axially-oriented outer diameter stress corrosion cracking (ODSCC). This particular form of SG tube degradation occurs entirely within the intersections of the SG tubes with the tube support plates (TSPs).

The present voltage values for the ODSCC repair criteria are based on the assumption of a ''free span'' exposure of the SG tube flaw; i.e., no credit is given for any constraint against burst or leakage, which may be provided by the presence of the TSPs. This approach is, in turn, based on the assumption that under postulated accident conditions, the TSPs may be displaced sufficiently by blowdown hydrodynamic loads such that a SG tube flaw which was fully confined within the thickness of the TSP prior to the accident would then be fully exposed. This approach was first advanced by the NRC staff in a draft generic letter issued on August 12, 1994, which was subsequently modified slightly and issued as Generic letter (GL) 95–05, ''Voltage-Based Repair Criteria For Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking,'' dated August 3, 1995. The previous license amendments related to the issue of ODSCC were based to a large extent on the draft generic letter cited above.

The fundamental difference between the pending proposal to raise the lower voltage repair limit to 3.0 volts and the methodology contained in GL 95–05, is that the licensee proposes to install certain modifications to the SG internal structures, thereby limiting to a small value, the maximum displacement of the TSPs under accident conditions. The proposed structural modifications consist of expanding a limited number of SG tubes only on the hot leg side of the TSP, at each of the intersections of the tubes with the TSPs. The purpose of this approach would be to greatly reduce the probability of SG tube burst under postulated accident conditions by several orders of magnitude. There would be a negligible impact on the primary-to-secondary SG tube leakage under accident conditions.

While the voltage-based repair criteria for ODSCC flaws are applicable only to Byron 1 and Braidwood 1, the pending request for license amendments involves all four units in that both stations have a common set of TSs.

Before issuance of the proposed license amendments, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendments would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The previously evaluated accidents of interest are steam generator tube burst and main steam line break [MSLB]. Their potential impact on public health and safety due to the change in SG tube plugging criteria proposed in this amendment request is very low as discussed below. Tube burst related to the types of cracks under